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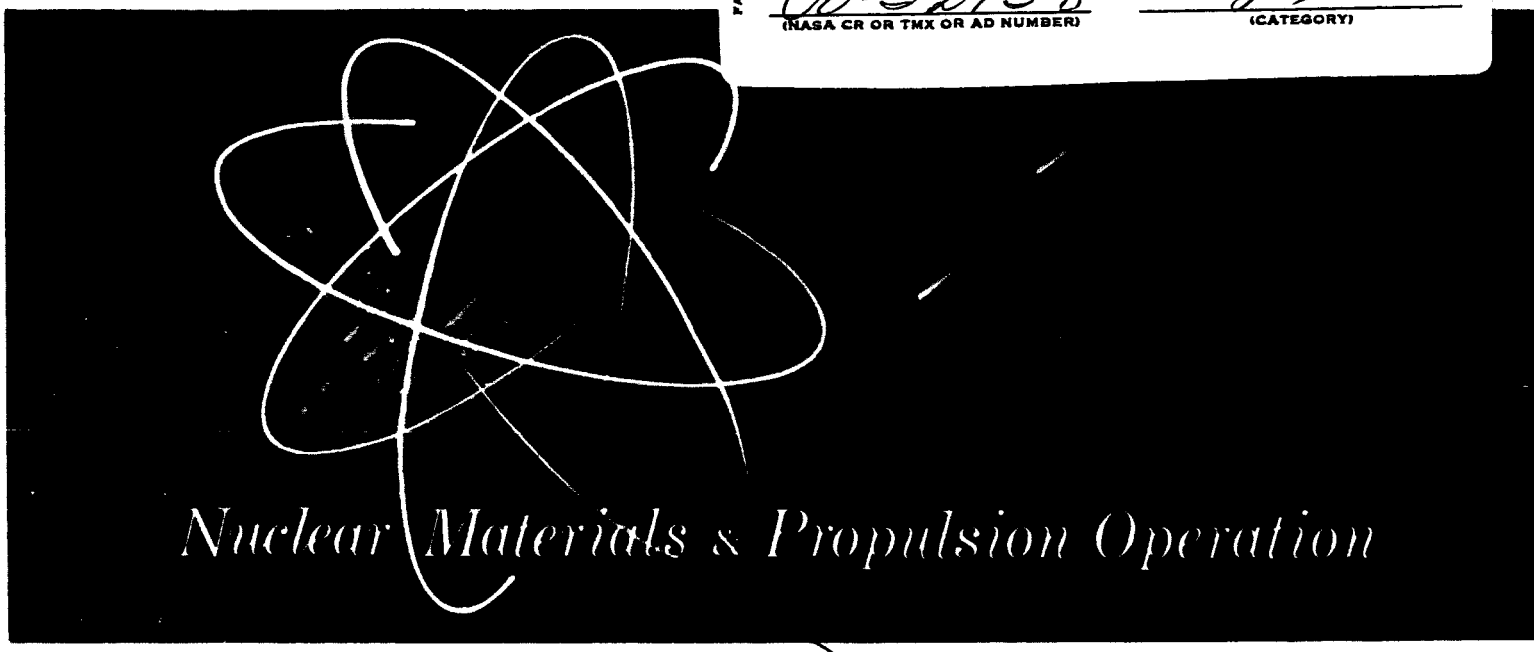

INTRODUCTION TO NUCLEAR PROPULSION

Lectures 5 and 6 - SHIELD PHYSICS

John Moteff

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INTRODUCTION TO NUCLEAR PROPULSION

Lectures 5 and 6 - SHIELD PHYSICS

John Moteff

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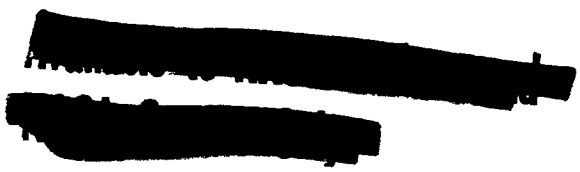

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PREFACE

Although shield technology was developed to a certain extent long before the operation of the first nuclear reactor, there still remain many areas that need further refinement. As a general rule, many of the interactions of radiation with shield materials are known to the extent necessary to generate a shield design based on approximate methods of analysis. The differential reaction cross sections, however, are not available at this time for many of the materials of interest for the routine computations of radiation penetration and shield heating using more sophisticated analytical techniques. The use of Monte Carlo programs have helped in the analysis of shields of complex geometries, but these techniques are not available to every shield designer and should not be used for the early phases or the preliminary shield designs.

The material of Shield Physics covered in lectures 5 and 6 of the course on Introduction to Nuclear Propulsion should give the reader who is not familiar with shielding technology an indication of the basic concepts that he may encounter in his work in nuclear propulsion design. Recognizing these concepts should help the reader communicate better with the shield experts and to request specific analysis so that the contribution to the overall design would be the best possible.

It is important to recognize that the one important output that a propulsion engineer requires from the nuclear shield engineer are the radiation levels within the shield and also those external to the reactor-shield assembly. Based on this information he may consult the materials engineer regarding radiation damage, induced activity or the thermodynamics and stress engineer regarding materials integrity.

Isodose patterns such as shown for the NERVA engine (Figure P1) are common figures of the type that appear in all design specifications of nuclear propulsion systems. Based on such patterns, either given as isodose or isoflux lines about an operating or shutdown reactor, the optimization of the overall nuclear vehicle system can be made.

Some of the radiation problem areas external to the reactor-shield assembly have been discussed in a recent article on the RIFT program by Col. W. C. Fellows in the recent issue of Astronautics (December 1962). Col. Fellows indicates the sources of radiation that would be encountered by a nuclear powered vehicle when tested on the ground, in the air and off into space. The vehicle structure and equipment will also perturb the radiation patterns shown in Figure P1. The structure scattering, air scattering and radiation heating mentioned by Col. Fellows are illustrated in Figure P2.

These lecture notes are based to a large extent on past experience in the GE-ANP program and reflect in several cases work generated by Gunnar Thornton and W. E. Edwards either in discussions or through their various reports.

As a further guide to the reader who may be required to dig deeper into shield theory, experiment or design either by choice or by job assignment, the following references should be reviewed and studied.

Reactor Handbook, Second Edition
Volume III, Part B, Shielding
Blizard and Abbot
Interscience Publishers
1962

Fundamental Aspects of Reactor Shielding
Goldstein
Addison-Wesley Publishing Company, Inc.
1959

Radiation Shielding
Price, Horton and Spinney
Pergamon Press
1957

Reactor Shielding Design Manual
Rockwell
TID-7004
(1956)

Principles of Nuclear Reactor Engineering
Glasstone
Van Nostrand
1955

Introduction to Nuclear Engineering
Stephenson
McGraw-Hill Book Company, Inc.
1954

Gas-Cooled High-Temperature Nuclear Reactor
Design Technology. Part E. Shield Design
APEX-800, Part E
FPLD
General Electric Company
1962

FIGURE P1

OPERATIONAL RADIATION LEVELS FOR SHIELDED NERVA ENGINE

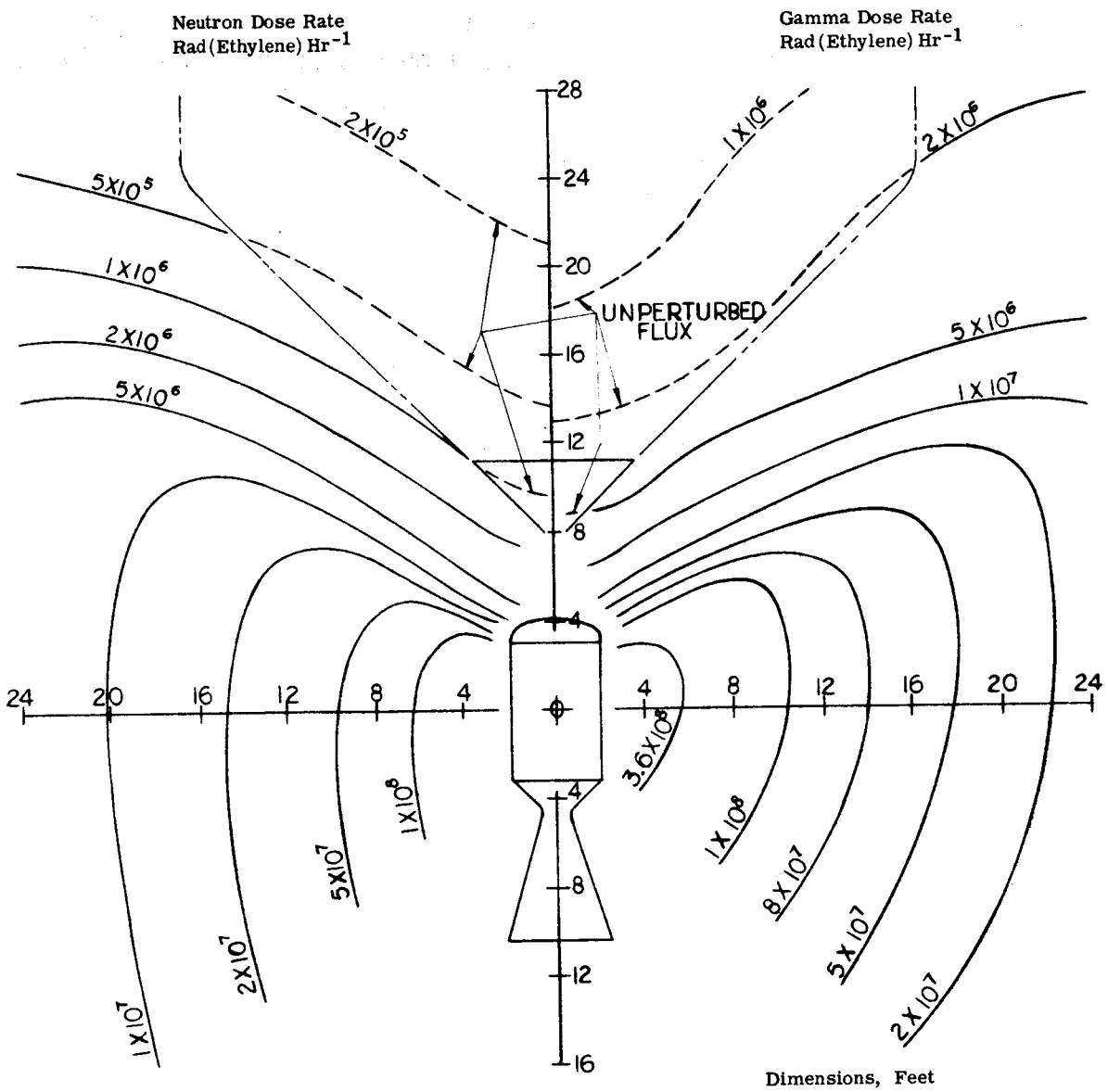
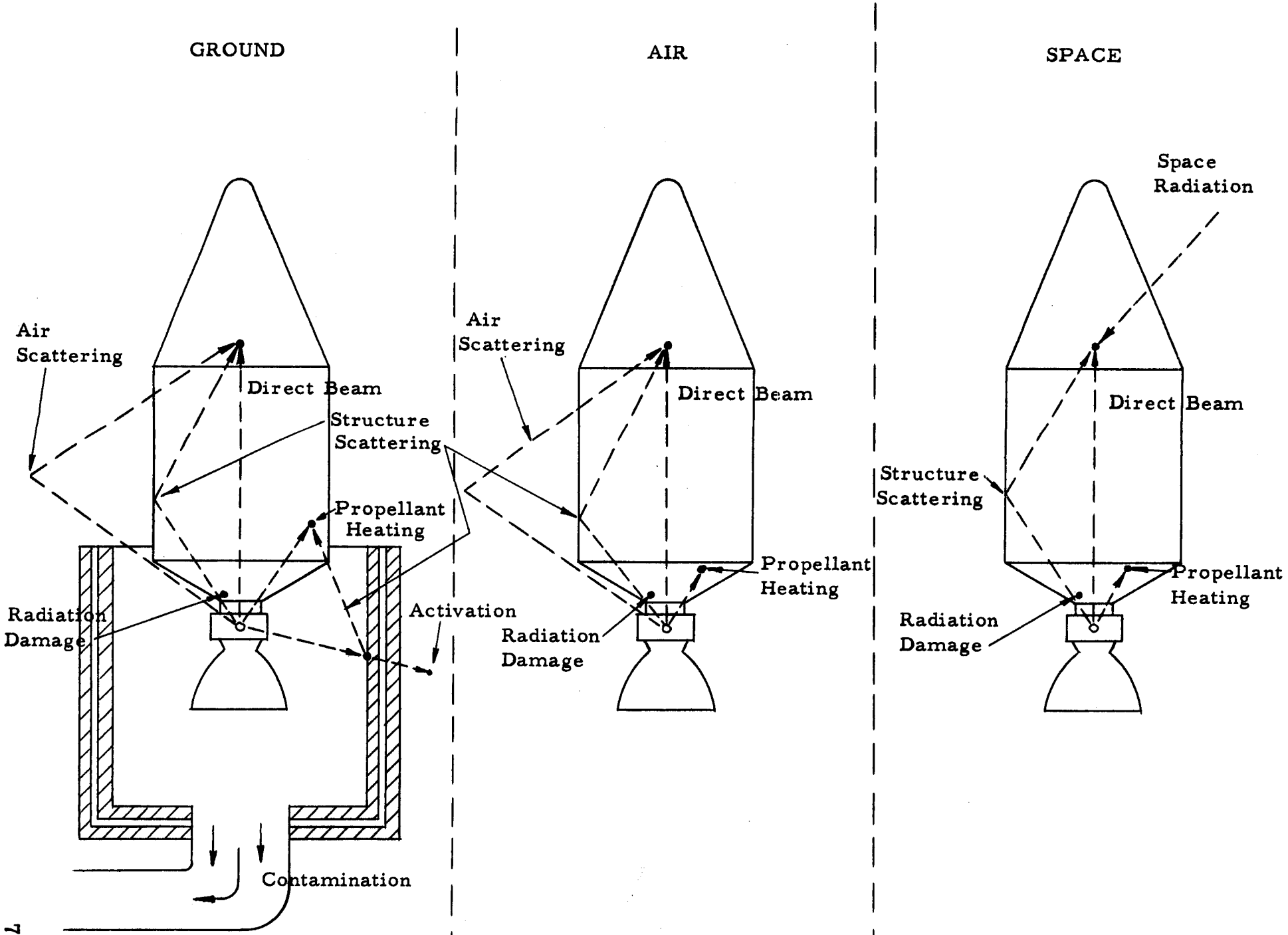


FIGURE P2 RADIATION CONSIDERATIONS FOR NUCLEAR VEHICLES



1.0 INTRODUCTION

A shield is needed to protect personnel and equipment from the adverse effects of nuclear radiation. Although shielding technology was developed long before the operation of the first nuclear reactor, the application to high performance nuclear propulsion systems requires further refinements.

The approach to the problem of radiation protection in the vicinity of an operating nuclear power plant may be generalized in the following manner. A mass of radioactive or of fissioning material may affect an observer principally by any one of three methods or combinations of these methods. The radiation may (1) penetrate whatever shield material has been set up to protect the observer, (2) be reflected (scattered) from the surroundings, such as buildings, structures, or air, to an observer who is screened from the direct rays of the radioactive source, or (3) in turn induce activity in materials or equipment which are in the vicinity of the neutron environment and subsequently transported into the observer's vicinity. Of course, the length of time that the observer remains in the radioactive field also determines the total radiation received.

1. In the simplest case the source is considered to be monoenergetic and infinitely small for means of calculating the radiation penetration. The intensity of the radiation then decreases with the square of the distance between the source and the observer. This divergence is generally called the geometrical attenuation. If some shield material is interposed, the intensity decreases exponentially with the thickness of the shield. This effect is generally called material attenuation. These conditions, however, are seldom realized in practice and a working calculation must therefore take into account: (a) the finite dimensions of the source, (b) the energy spectrum of the source, (c) the effect of the absorption of the radiation within the source itself, and (d) deviations from the idealized exponential absorption law.
2. If the shielding material does not completely surround the radioactive source, the radiation may scatter or reflect, from various objects along its line of sight, back to the position of the observer. Coolant ducts, instrument channels and other voids in the shield material may be sources of scattered radiation. In many cases secondary shielding may be placed, to protect the observer, either in the shield region or in the vicinity of the observer.
3. The activity from material or equipment that is exposed to neutron radiation may be calculated if the cross section for activation, the conditions of exposure and the nature of the induced activity are

known. These radioactive materials such as equipment, fuel elements, test specimens will require shielding when they are removed from the neutron environment and placed in the vicinity of the observer. If the radioactivity is too high or the time for decay is very long, the equipment may be discarded by burial or may be maintained by means of remote tools and the use of hot cells.

The first step in the design of any shield is the definition of the radioactive source. The necessary nuclear quantities are the spatial and the energy distribution of all sources of neutrons and gamma rays. Accurate definition of these quantities requires detailed knowledge of the distribution of the fission, neutron capture, and any elastic or inelastic scattering within the reactor-reflector assembly. Only a refined reactor analysis can develop this information.

Ordinarily, a power density distribution is available from the preliminary design reactor engineer. A first assumption is that the distribution of all sources is the same as the power density distribution. This source configuration may be approximated by a combination of a cosine and an exponential function. In fact the assumption of a flat source distribution may be entirely adequate in the first shield analysis. After the power level is defined, the fission rate may be established.

1.1 SOURCES OF RADIATION PARTICLES

Radiation from nuclear reactors which are important to shield designers are summarized in Figure 1. In principal, the radiation which might escape a reactor system include alpha and beta particles, gamma rays, neutrons, fission fragments, and even protons resulting from the (n, p) reaction. The energy of these particles will vary considerably. As far as the shield design is concerned, however, only gamma rays and neutrons need be considered. These radiations are by far the most penetrating. For instance, any material which could attenuate the gamma rays and neutrons to a sufficient extent will automatically reduce all the others to negligible proportion.

More than 99 percent of the neutrons which are liberated in fission, are released promptly with energies ranging from about 0.5 Mev to approximately 14 Mev and higher. These fast neutrons are slowed down as a result of elastic collision with a moderator or by inelastic collision with nuclei of heavier elements and some will escape from the active core of the reactor. Some of the neutrons, however, will be subsequently absorbed within the reactor system. In the case of inelastic scattering of neutrons, the excited target nucleus will release its excess energy in the form of gamma radiation. On slowing down many neutrons will be absorbed,

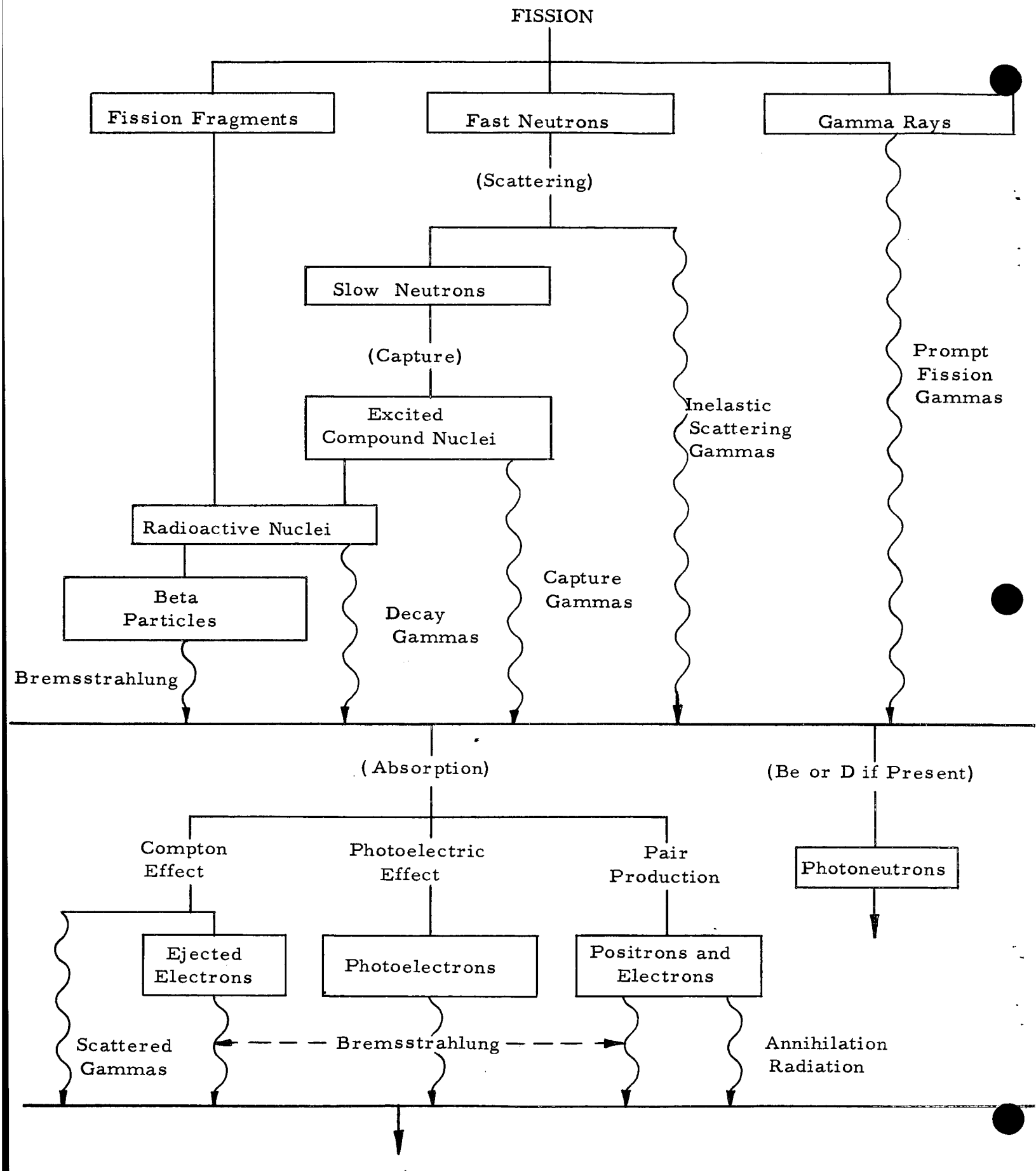


Figure 1. Radiations from nuclear reactor

releasing gamma radiation. This reaction is generally called the (n, γ) or capture gamma reaction. In addition to capture by moderating material and structural material, it must be remembered that both uranium and other fuels will absorb neutrons as a non-fission or parasitic capture.

If beryllium or deuterium is present in the reactor, the action of gamma rays, if of sufficient high energy, will produce photo-neutrons. This photo-neutron reaction is designated as a (γ, n) reaction.

The interaction of the various gamma photons with material will produce photoelectrons, Compton electrons, and for energies in excess of 1.02 Mev, position-electron pairs. As these electrons are slowed down, Bremsstrahlung will be produced, while the recombination of electrons and positrons will result in the formation of annihilation radiation of energy 0.51 Mev.

By this time it should be obvious that nuclear radiation escaping from a reactor will be very complex in nature. In addition when the neutrons and gamma radiations of various energies enter the shield, they will suffer interactions of the same type as those that occur within the reactor. Therefore, throughout the shield there will be produced secondary gamma radiations due to inelastic scattering, capture and other processes involving nuclei constituting the shielding material. Unless proper precautions are taken in the design of the shield, the secondary components of the radiation could lead to serious design problems. For example, if a slow neutron were captured near the outside of the shield, it could give rise to capture gamma radiation of high energy which might emerge from the shield with little or no energy loss.

2.0 SUMMARY OF BASIC RADIATION PHYSICS

2.1 GAMMA RAY CROSS SECTION

Although there are approximately seven types of reactions that could influence the penetration of gamma photons (gamma rays) through matter, only three reactions are generally used in most of the routine shielding calculations. The interaction of these gamma photons with matter may result in complete absorption or they may be degraded in energy and deflected from its original path. The effectiveness of a shield in attenuating gamma rays is, therefore, determined by the combined effect of several of these processes occurring at the same time or successively. In general, the photon energies of interest to the shield designer are between 0.1 and 10 Mev.

2.1.1 Photoelectric Effect

In the photoelectric effect a gamma ray photon with energy greater than the ionizing energy of an orbital electron in an atom, interacts with the latter in such a way that the whole of the gamma ray energy is transferred to the electron which is consequently ejected from the atom. The photoelectron, as it is called, behaves like a beta particle in its passage through matter. The extent of the photoelectric interaction depends on both the energy, E, of the gamma radiation and the atomic number, Z, of the absorbing material. As a rough approximation,

$$\text{Probability of photoelectric interaction} \approx \text{constant} \times \frac{Z^n}{E^3} \qquad (1)$$

where n varies from 3 for gamma rays of low energy to 5 for gamma rays of high energy. It is apparent, therefore, that the photoelectric effect increases with increasing atomic number of the absorber and with decreasing energy of the gamma rays. Following the expulsion of the photoelectron, another electron, from an outer orbit, takes its place in the atom; this transition is accompanied by the emission of characteristic X-rays.

2.1.2 Compton Scattering

In Compton scattering a gamma ray photon makes an elastic (or billiard ball) collision with an outer electron of an atom of the absorbing material. Such an electron is loosely bound so that it behaves as if it were completely free. In the collision both momentum and energy are conserved, and part of the incident photon is transferred to the electron; at the same time the photon is deflected, that is, scattered from its initial path (see

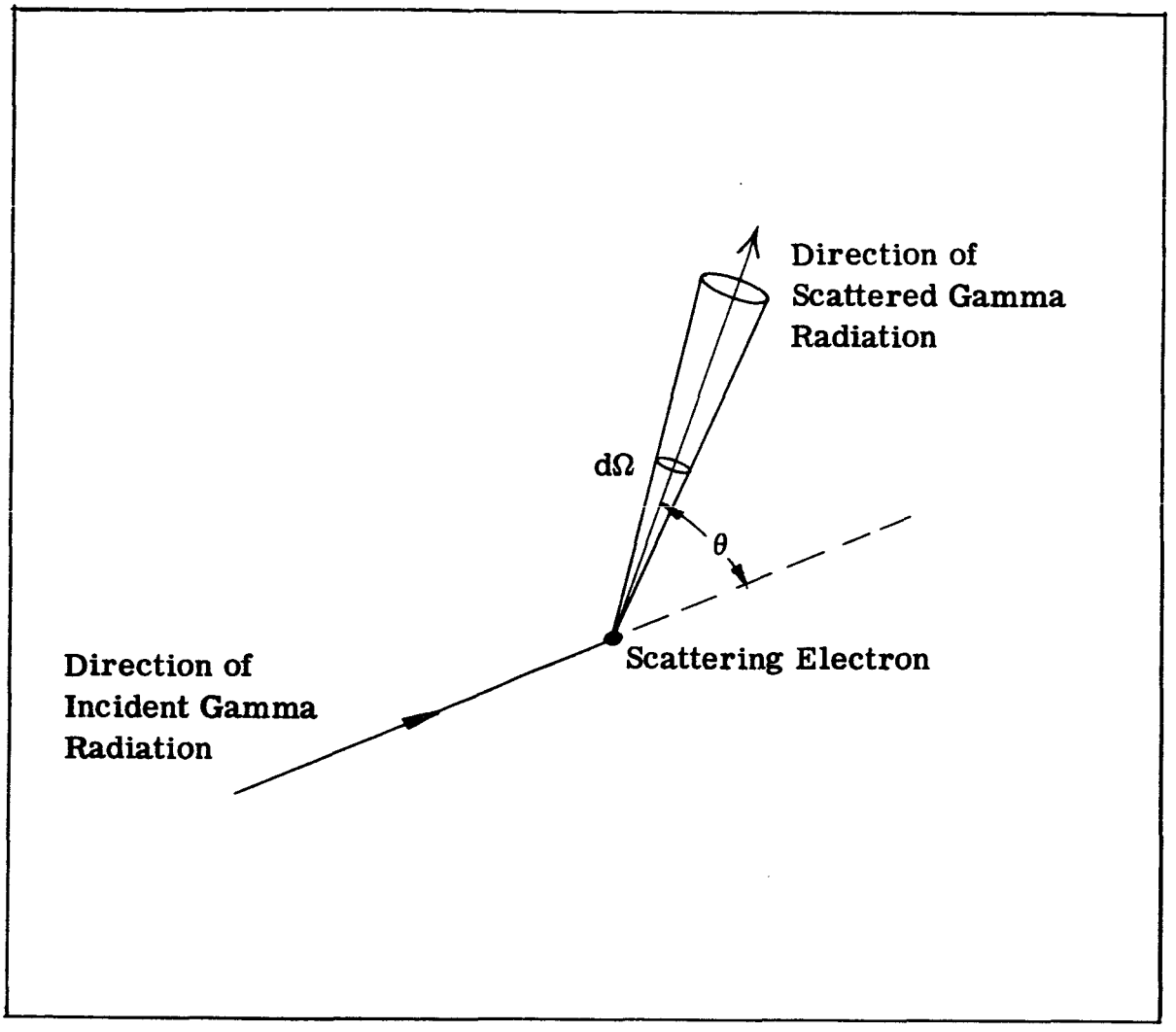


Figure 2. Gamma Photon Scattering

Figure 2). The relation between the energy E of the incident photon, E' of the scattered photon, both in Mev, and the scattering angle θ, is given by

$$E' = \frac{0.51}{1 - \cos\theta + \frac{0.51}{E}} \tag{2}$$

If the scattering angle is small, $\cos\theta \cong 1$ and then E' is approximately equal to E. This means that the scattered photons with energy close to the energy of the incident photon, proceed in a nearly forward direction. On the other hand, for $\theta = 90^{o}$, $\cos\theta = 0$ and then E' is equal

$$E' = \frac{0.51\,E}{E + 0.51} < 0.51 \text{ Mev} \tag{3}$$

Consequently a photon scattered at right angles can not have energy greater than 0.51 Mev. This effect is very important in the calculation of the shielding required to protect personnel or equipment from gamma radiation which is scattered from an air medium, or as in the case of a space propulsion, from the vehicle structure.

The fraction of the initial energy carried by the scattered photon for different scattering angles may be obtained from Equation 4.

$$\frac{E'}{E} = \frac{0.51}{E(1 - \cos\theta) + 0.51} \tag{4}$$

For a specified value of θ the fraction decreases with decreasing energy of the incident photon. In other words for a given scattering angle, the greater the energy of the incident photon, the smaller the fraction remaining on the scattered photon, and the larger the fraction of energy loss by the gamma ray in the Compton interaction.

Since the Compton effect involves interaction between a photon and an electron, its magnitude is dependent upon the number of orbital electrons in the atom of the absorber; this is the same as the atomic number. The Compton interaction is, therefore, directly proportional to the atomic number of the absorber so that, like the photoelectric effect, it is more significant with materials of high atomic number. The energy dependence of the Compton effect is given by the Klein-Nishina formula. This is somewhat too complicated to reproduce here but it can be stated that the Compton interaction decreases monotonically with increasing energy of the gamma radiation. As a rough approximation, it is possible to write

$$\text{Probability of Compton interaction} \approx \text{constant} \times \frac{Z}{E} \qquad (5)$$

The significant difference between the photoelectric and Compton effects is that the photoelectric effect is a true absorption process; that is to say the photon is absorbed as stated above. In the Compton process, however, there is merely a decrease in photon energy, the extent of this decrease being greater the larger both the initial energy and the scattering angle. A photon which has been scattered in a Compton interaction will thus still be present in the medium, although it will have a somewhat lower energy and may be moving in different directions.

2.1.3 Pair Production

In the pair production reaction, a gamma photon with energy in excess from 1.02 Mev passes near the nucleus of an atom, and the photon can be annihilated in a strong electrical field with the formation of the electron-positron pair. Since the energy equivalent of the total mass of an electron and positron is 1.02 Mev, this is the minimum energy necessary for the production of the pair of particles. Any energy of the gamma ray photon in excess of 1.02 Mev appears mainly as kinetic energy of electron and positron, with a small fraction transferred to the atomic nucleus. The particles produced tend to travel in a forward direction, the effect becoming more evident with increase in gamma ray energy.

The extent of pair production by gamma radiation of energy E (in Mev) is related to the atomic number Z of the absorber by

$$\text{Probability of pair production} \approx \text{constant} \times Z^2(E - 1.02) \qquad (6)$$

so that it increases with the atomic number with the absorbing material and with increasing photon energy in excess of 1.02 Mev. For absorbers of high atomic number, it becomes a dominant type of interaction for gamma rays for energies in excess of about 5 Mev.

As in the photoelectric effect, pair production results in absorption of gamma rays photons. It is true that some of the positrons formed will be neutralized by the electrons nearby and produce annihilation radiation, consisting in general of two 0.51 Mev photons.

The gamma ray total mass absorption cross sections for a few of the common shielding materials are given in Figure 3. Figure 4 shows the energy absorption cross sections for the same materials. The difference between the energy absorption and the total cross sections is due to the

Compton process which removes the photon from the beam by scattering, but does not remove all of its energy in that scattering event. An indication of the relative importance of the three types of gamma ray interaction with matter are shown in Figure 5 for the case of a low Z and a high Z material. Figure 6 is a plot of the total linear absorption cross sections for concrete and various window glasses used in reactor test facility designs.

2.2 NEUTRON CROSS SECTIONS

A high performance mobile reactor shield must absorb both the neutron and the gamma photons with the minimum thickness and weight possible. Since very little direct capture of fast neutrons occur, the removal of fast neutrons in the shield will, therefore, depend mainly on slowing down (moderation) to thermal energies where they are subsequently captured. For the attenuation of moderately fast neutrons (0.1 to 5 Mev) it is practically essential that the shielding material contain some low Z material such as beryllium, lithium, or hydrogen. The loss of energy per collision is greater with hydrogen, since the fractional energy loss of a neutron on collision with a nucleus of atomic weight A is proportional to $A/(A + 1)^2$, which is a maximum for $A = 1$. However, for neutrons above about 5 Mev a high Z material may result in a better shield based on a thickness criteria, due to the higher inelastic scattering cross section. The choice of material will depend therefore on the design objectives of the shield system.

As a general rule, after a fast neutron makes its first collision in a hydrogeneous shield, its motion can be described as a diffusion in which the net forward progress through this shield is relatively slight. The diffusion usually ends with thermalization and capture of the neutron in the vicinity of its first collision. In a nonhydrogeneous engineered shield material, the capture may take place at some distance from the initial collision. This is not true,for instance, for high purity carbon which is used as a thermal neutron column for transporting a copious quantity of neutrons out of a reactor to some experimental apparatus. In many cases, a poison such as boron or lithium, which have a large thermal neutron absorption cross section, is added to the shield material to increase the captures within the shield. In any event a neutron shield material must be able to reduce the intensity of the fast neutron flux as it penetrates the shield. Since collisions will reduce the neutron energy, the scattering or absorption rate will also vary since the material will have different reaction cross sections for the different energies. The knowledge of the cross sections are of considerable importance in the calculation of the final removal of neutrons in a shield.

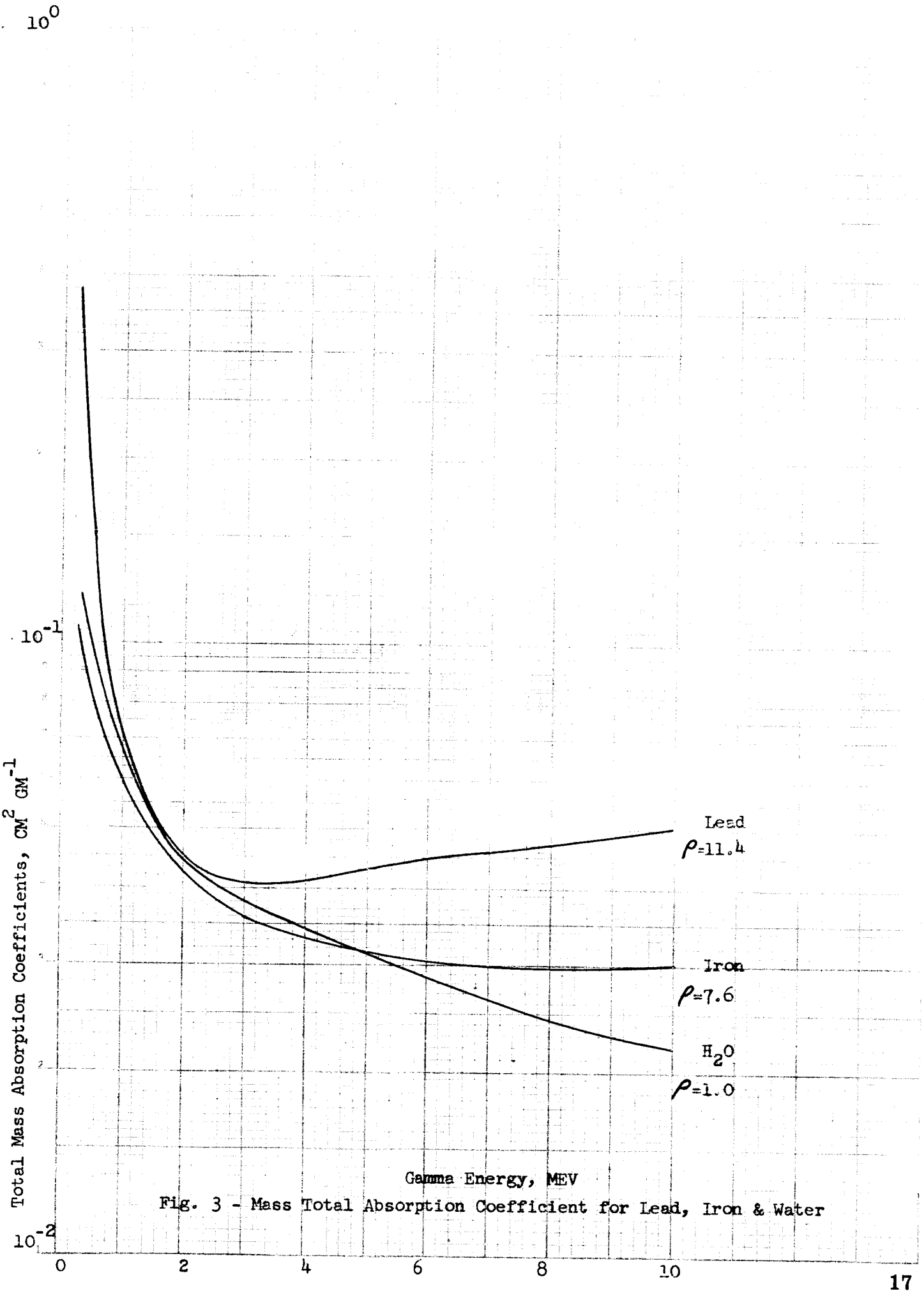


Fig. 3 - Mass Total Absorption Coefficient for Lead, Iron & Water

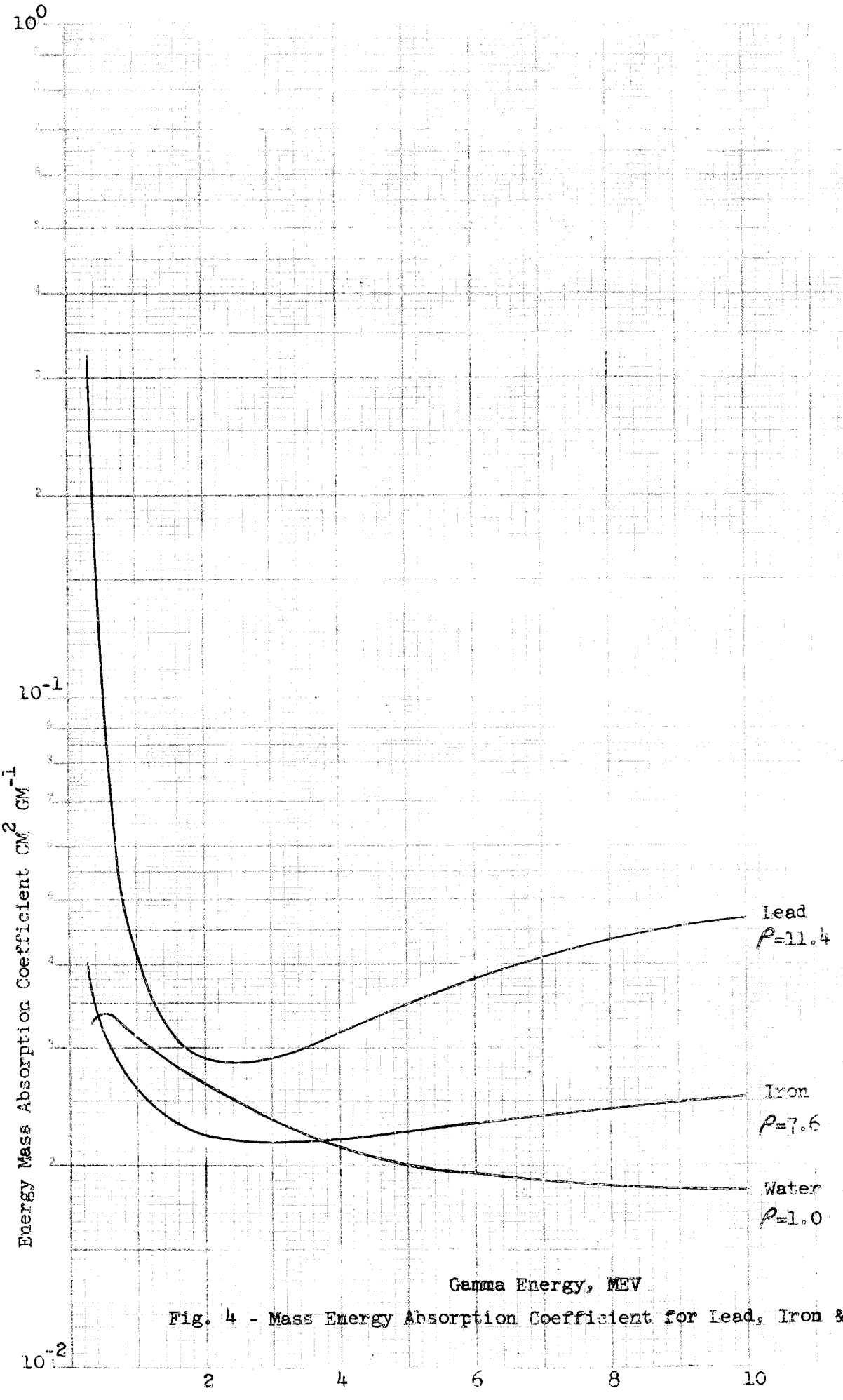


Fig. 4 - Mass Energy Absorption Coefficient for Lead, Iron & Water

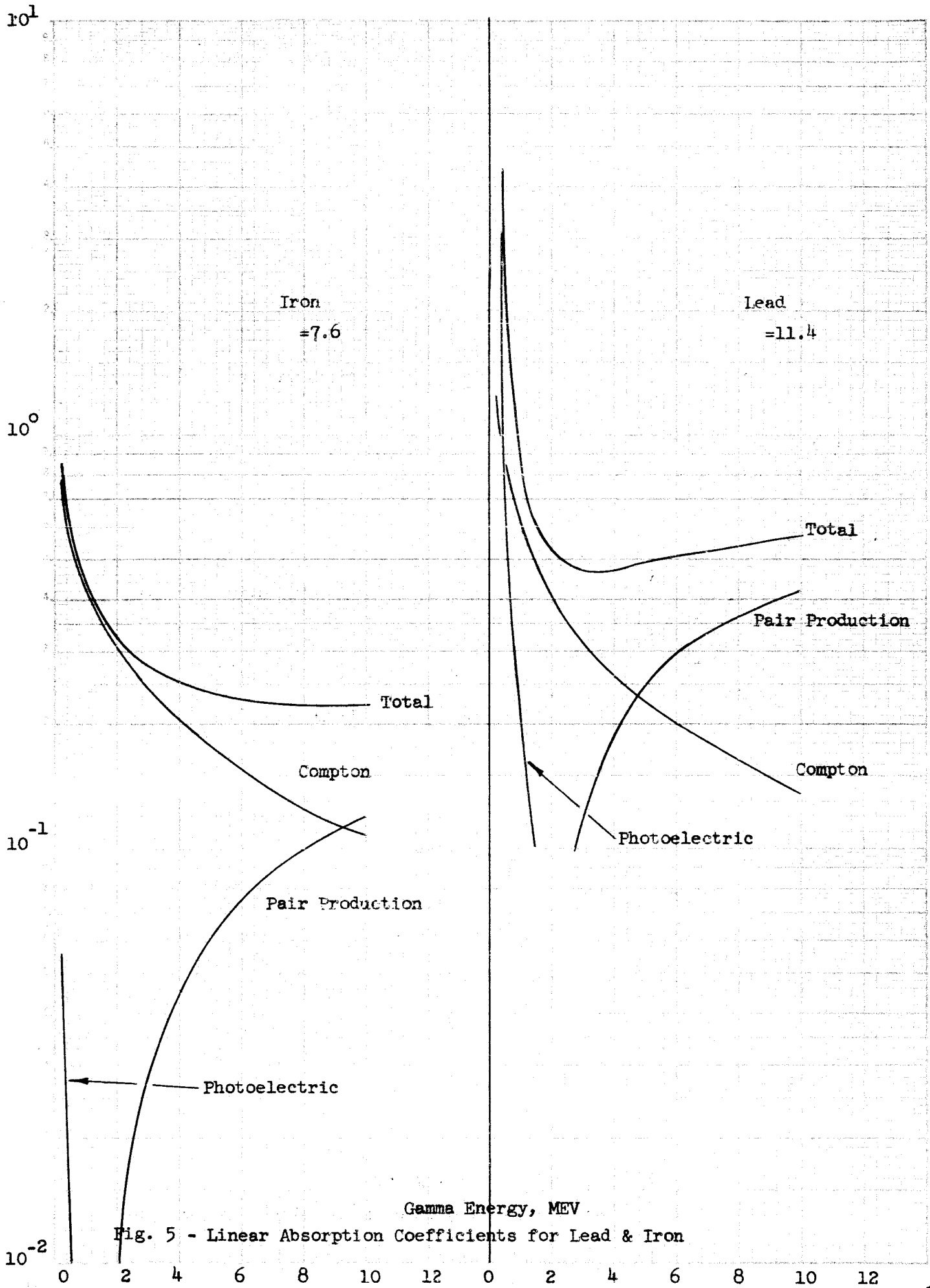


Fig. 5 - Linear Absorption Coefficients for Lead & Iron

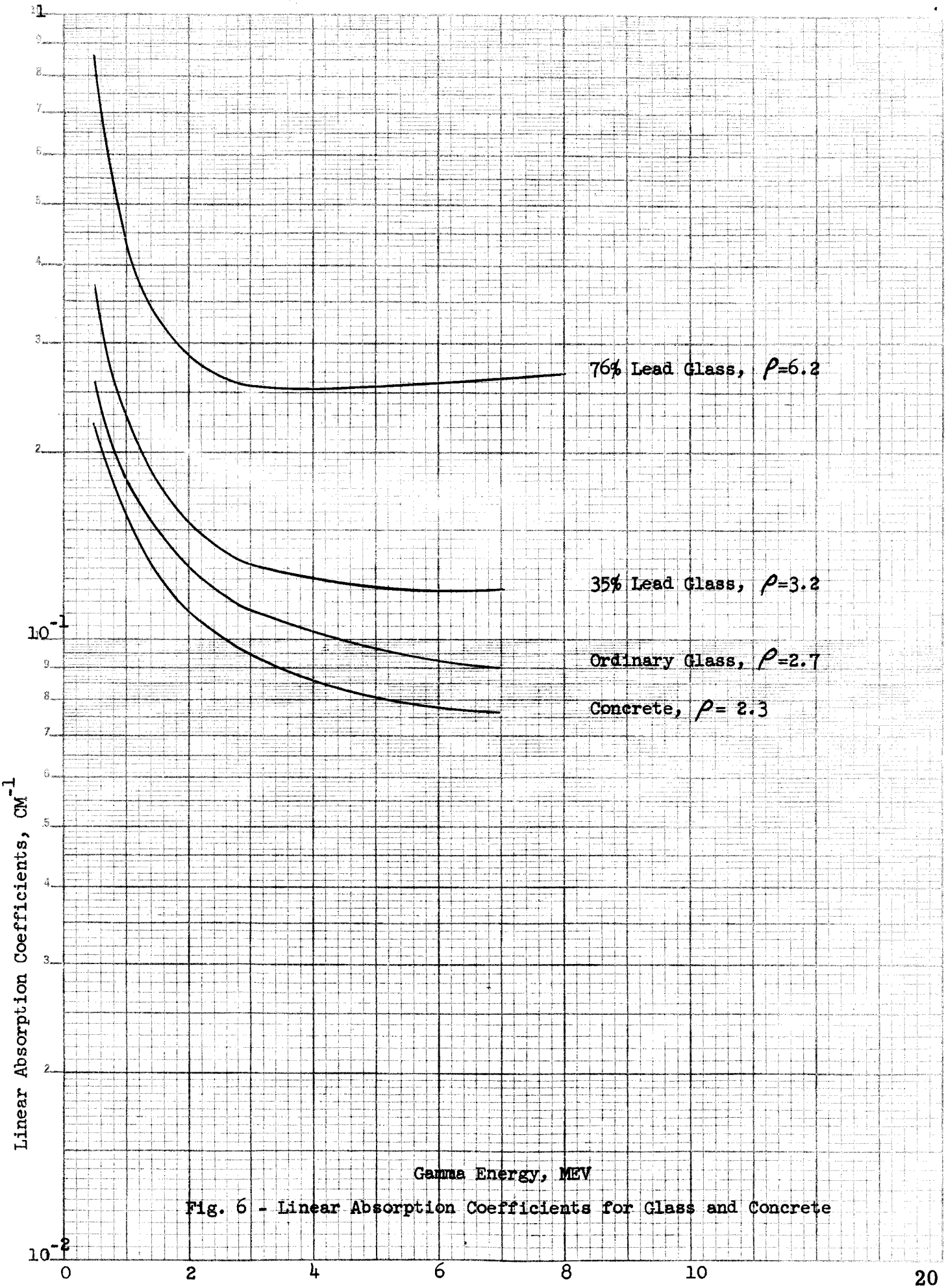


Fig. 6 - Linear Absorption Coefficients for Glass and Concrete

2.2.1 Total Cross Sections

Figure 7 shows the variation of the total neutron cross section for iron and water. The effect of the hydrogen in the water becomes obvious for neutron energies below about 1 Mev. The resonance structure of the water cross section above 1 Mev is due to those present in the oxygen. Note the dip in the total cross section of iron at about 24 Kev. This type of window in cross sections will essentially pipe through (transport) the medium a large number of neutrons of that energy, 24 Kev for the case of iron. Adding elements, such as chromium, manganese, nickel or others, to the iron will essentially cover the window in the cross section so that the neutrons will not leak out at that energy. This phenomena has been experimentally verified.

2.2.2 Removal Cross Sections

Fast neutron removal cross sections have been experimentally determined at the Lid Tank Facility at Oak Ridge National Laboratory. The technique is to measure the neutron dose in a water medium with and without a slab of shield material interposed between the source of fission neutrons and the neutron detector. These empirical cross sections corrects, in part, the multiple scattering of neutrons in the shield slab and may be used to calculate the exponential attenuation through the material. The use of these cross sections in the calculation assumes that the shield material is backed by at least a few inches of some hydrogenous material.

Another cross section that is generally used for the calculation of fast neutron penetration through shield material is the energy dependent fast neutron collimated removal cross section. This cross section is defined as

$$\Sigma_R(E) = \Sigma(E)_{\text{absorption}} + \Sigma(E)_{\text{inelastic}} + \Sigma(E)_{\text{elastic}} [1 - \bar{\mu}(E)] \quad (7)$$

where $\bar{\mu}(E)$ is the average cosine of the angle of elastic scattering in the laboratory system at energy E.

The fast neutron energy dependent collimated removal cross section, however, does not take into account the slowing down and migration of the neutron below several Mev. The flux intensity of high energy neutrons that have penetrated to some distance within the shield will, however, be the source of fast neutrons, at their next collision, for a subsequent neutron diffusion calculation which is used to determine the energy and spacial distribution of neutrons within the shield. A few of these cross sections are plotted in Figure 8.

Figure 7 - Total Neutron Cross Section for Iron and Water

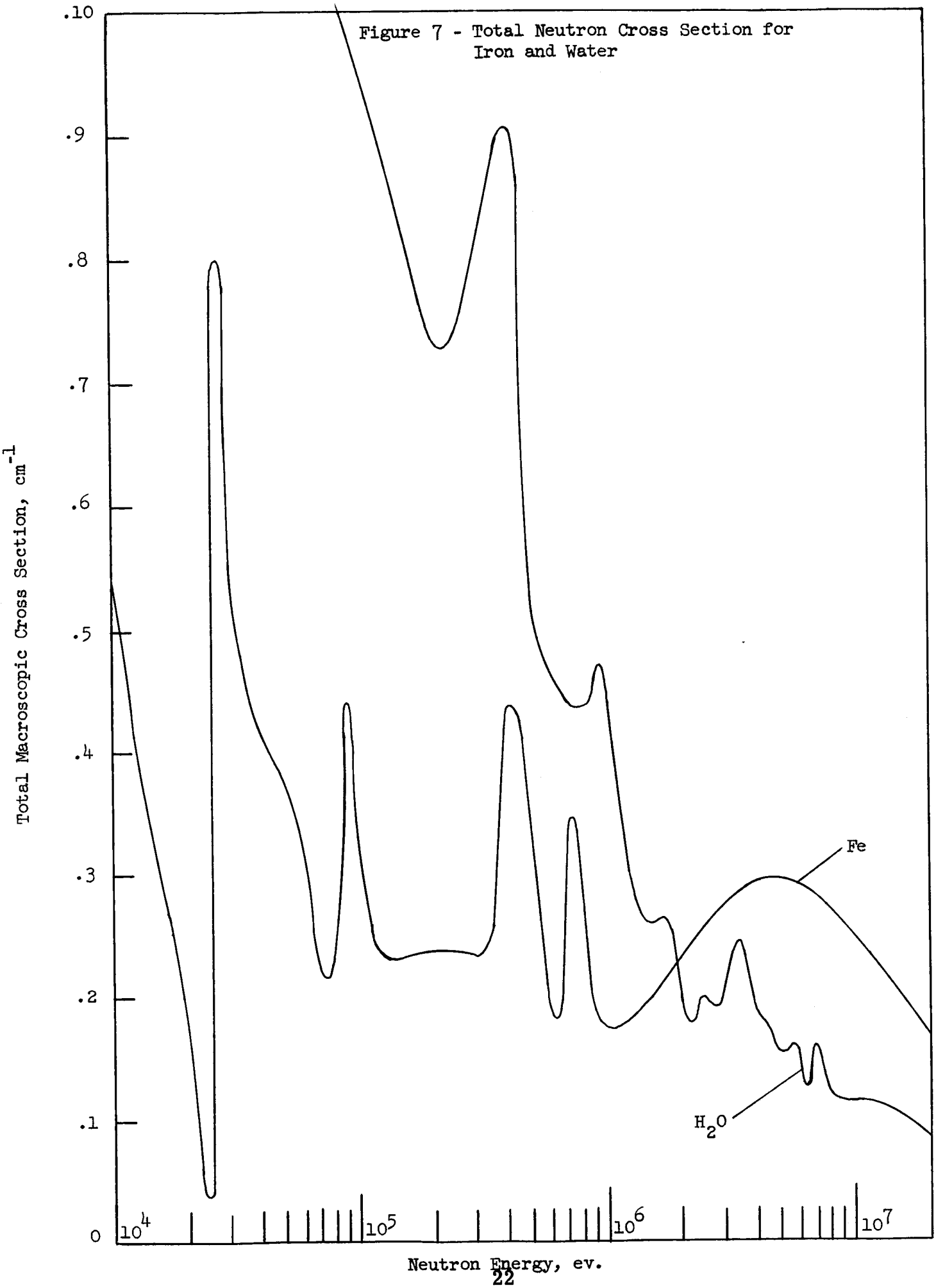
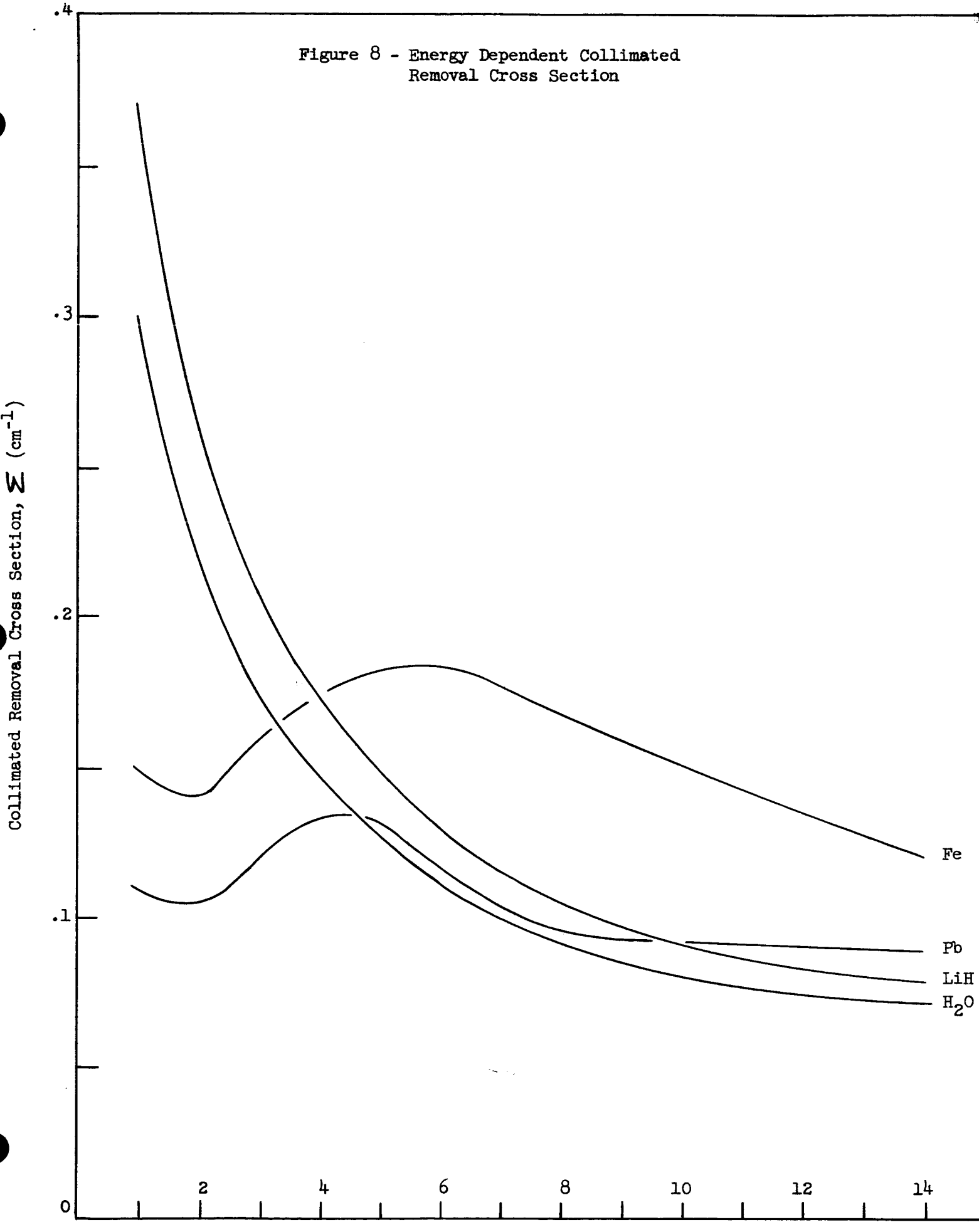


Figure 8 - Energy Dependent Collimated Removal Cross Section



2.2.3 Activation Cross Sections

Structural and component materials are usually alloys or mixtures containing several different chemical elements. Each chemical element in turn exists in the form of several different natural isotopes each of which has the same atomic number Z but a different atomic weight A. An isotope is designated by the symbol $_{Z}(\ )^{A}$ For example, the symbol $_{25}Fe^{54}$ designates the naturally occurring iron isotope of atomic weight 54. Since the atomic number of iron is always 25, the symbol may be abbreviated to Fe^{54}.

A number of different type reactions are involved in neutron absorption in each isotope of a chemical element. For example, the following reactions can occur when a neutron is absorbed in the isotope Fe^{54}.

Type Reaction	Designation
(n, p)	Fe^{54} (n, p) Mn^{54}
(n, α)	Fe^{54} (n, α) Cr^{51}
(n, γ)	Fe^{54} (n, γ) Fe^{55}
(n, 2n)	Fe^{54} (n, 2n) Fe^{53}

The gammas, protons, etc., produced in these reactions are emitted promptly, but the resulting product isotopes such as Mn^{54} are usually radioactive in the sense that they later emit decay radiation such as beta particles, and, in most cases, additional gamma photons.

The same radioactive isotope can be produced in several different materials due to different types of reaction. For example, the radioactive isotope Fe^{59} can be produced in the following reactions

$$Fe^{58}\,(n, \gamma)\,Fe^{59}$$

$$Co^{60}\,(n, p)\,Fe^{59}$$

$$Ni^{62}\,(n, \alpha)\,Fe^{59}$$

As a general rule most of the (n, p), (n, α) and (n, 2n) reactions are possible with neutron energies above several Mev. The (n, γ) reaction is due primarily to thermal neutrons and in many isotopes to epithermal neutrons. For induced activity calculations it is convenient to consider the neutrons to be in three energy regions; thermal, epithermal, and fast, defined as:

Region	Neutron Energy Range
Thermal	Below 0.4 ev
Epithermal	0.4 ev to 0.1 Mev
Fast	Above 0.1 Mev

2.3 RELAXATION LENGTH

A quantity $\lambda = 1/\Sigma$ may be defined where λ is called the "relaxation length" for radiation in the shielding materials whose total cross section is Σ reactions cm^{-1}. Thus λ has the dimensions cms. The expression for exponential attenuation in terms of λ may be written as $e^{-\Sigma x} = e^{-x/\lambda}$. For a shielding thickness $x = \lambda$, then $e^{-\lambda/\lambda} = e^{-1} = 0.3675$. Thus the relaxation length λ is the thickness of material required to reduce the radiation intensity of a narrow beam by a factor of 1/e.

For a shielding thickness $x = 2.3\lambda$, then $e^{-\Sigma x} = e^{-2.3\lambda/\lambda} = e^{-2.3} = 0.1$. Thus the shield necessary to attenuate radiation of a narrow beam by a factor of 10 is just 2.3 relaxation lengths in thickness. A factor of 100 reduction is obtained with 4.6 relaxation lengths, a factor of 1000 by 6.9 relaxation lengths, etc. The quantity $\Sigma x (= x/\lambda)$ or for shields of many layers, the quantity $(\Sigma_1 x_1 + \Sigma_2 x_2 + \cdots)$ is called the relaxation length thickness of the shield.

The neutron and gamma ray relaxation lengths for some of the common shield materials are listed in Table 1. These values are based on the more penetrating high energy reactor radiations and should be used with care since the cross section will vary with the neutron energy which in turn varies with the depth of penetration within the shield. The average neutron energy increases with an increase in shield penetration an effect sometimes called spectrum hardening.

2.4 RADIATION UNITS

Photon and neutron radiation are generally described by one of two methods. In the more basic description of the radiation, the particle direction, intensity and energy at a given position in space is specified. This radiation field denoted the flux or current intensity (commonly shortened to flux or current) may be expressed in terms of numbers or energy. They may also be differential quantities, i.e., expressed as intensities per unit energy interval or they may be total quantities which are integrated over energy.

TABLE 1

RELAXATION LENGTHS*
FOR FAST NEUTRONS AND GAMMA RAYS

Material	Density, g/cm^3	Relaxation Length, cm	
		Fast Neutrons	Gamma Rays
Lithium Hydride	0.73	7.8	41
Water	1.0	10	30
Graphite	1.6	9	19
Beryllium	1.8	7.5	18
Beryllium Oxide	2.3	7.5	14
Concrete	2.3	11	15
Barytes Concrete	3.5	8	10
Iron	7.8	6	3.7
Lead	11.3	9	2.5
Uranium	18.7	5.6	1.4
Tungsten	19.4	5.7	1.5

*Based in part on the high energy radiation leaving a large reactor core.

Because of the difficulty of measuring flux or current, especially in differential form, a second method of specifying radiation intensity is often used. In this method the absorption in an absorber, or the response of the absorber in the radiation field, is specified. Dose has probably been the most commonly used quantity of this type, but much confusion has arisen because of the numerous definitions of dose.

The International Commission on Radiological Units and Measurements (ICRU) now recommends the use of absorbed dose. In the ICRU definition, absorbed dose of any ionizing radiation is the energy imparted to matter, dose rate is the absorbed dose per unit of time. The unit of absorbed dose is the rad, defined as 100 ergs per gram of absorbing material. The material must always be specified. The ICRU also defines an exposure dose. The exposure dose of X- or gamma radiation at a certain place is a measure of the ability of the radiation to produce ionization. The unit of exposure dose of X- or gamma radiation is the roentgen (r). One r is an exposure dose of X- or gamma radiation such that the associated corpuscular emission per 0.001293 gram of air produces, in air, ions carrying one electrostatic unit of quantity of electricity of either sign.

Biological damage is not determined by absorbed dose alone since certain types of radiation are more effective in causing damage than others. Hence, it has been necessary to assign agreed relative biological effectiveness (RBE) factors to the different types of radiation so that RBE doses can be computed. Relative biological effectiveness has been defined as the biological potency of one radiation as compared with another. It is numerically equal to the inverse of the ratio of absorbed doses of the two radiations required to produce equal biological effect. The standard of reference is usually 200 kilovolts of X-radiation, which thus has an RBE of 1. The RBE for X-rays, electrons, and positrons of any specific ionization is 1 (the RBE for gamma rays is also 1.). Values of the RBE of neutrons as a function of energy are plotted in Figure 9. If the neutron energy spectrum is unknown, an RBE of 10 should be assumed.

The unit used in the description of radiobiological effects on man is the rem. The dose in rems is numerically equal to the product of the absorbed dose in rads (tissue) and the value of the RBE applicable for the radiation in question. Thus,

$$\text{RBE dose (in rems)} = [\text{RBE}]\,[\text{absorbed dose in rads (tissue)}]$$

The total RBE dose of mixed radiations is assumed to equal the sum of the products of the absorbed dose of each radiation and its RBE.

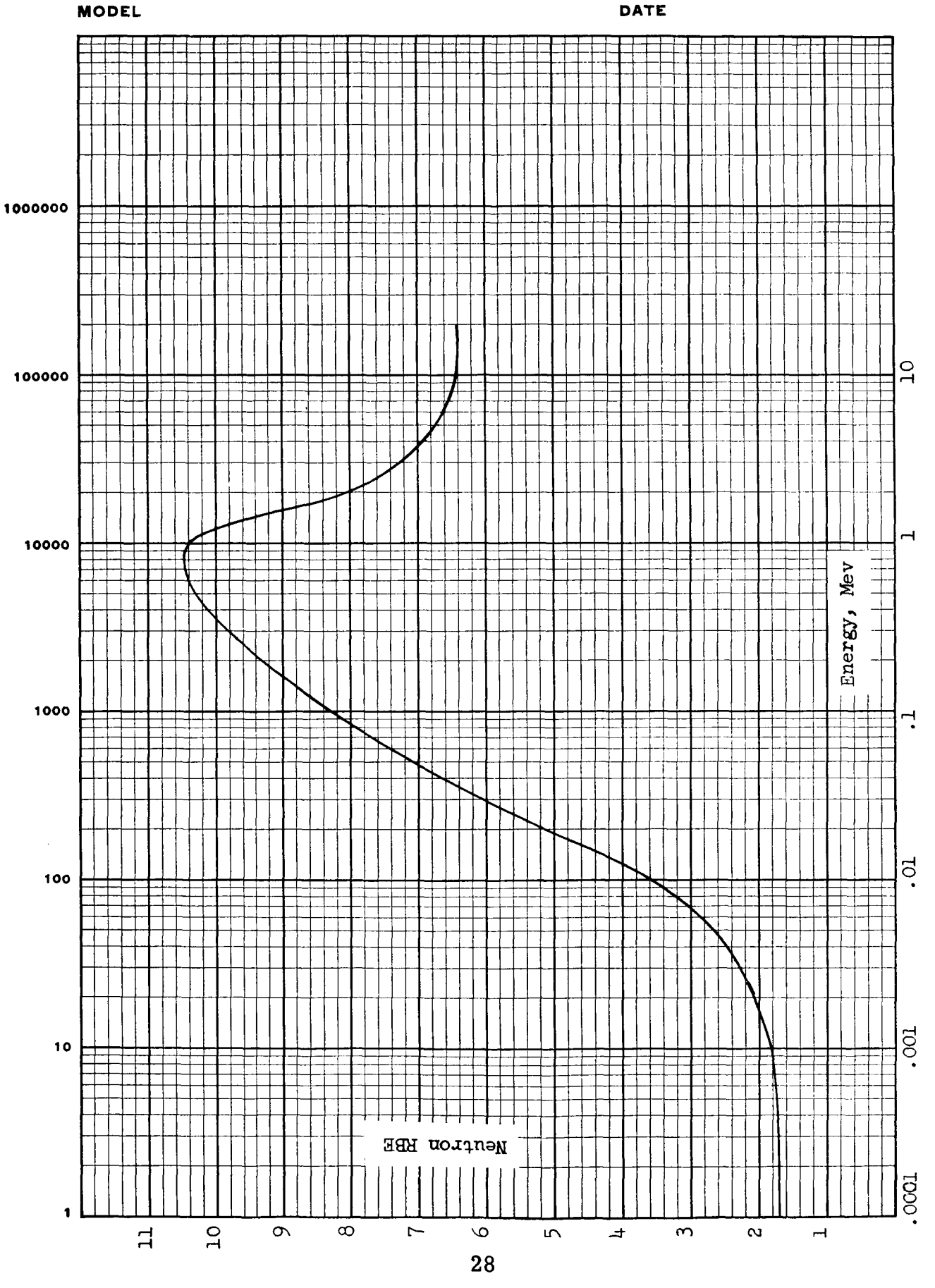


FIGURE 9

MAXIMUM NEUTRON RELATIVE BIOLOGICAL EFFECTIVENESS

Maximum permissible radiation exposures to man are specified in reference and an addendum dated April 15, 1958. For orientation purposes, the following are summaries of the basic rules for external exposure:

1. Accumulated dose (radiation workers)
   a. External exposure to the most critical organs. Whole body, head and trunk, active blood-forming organs, or gonads: the maximum permissible dose (MPD) accumulated at any age, shall not exceed 5 rems multiplied by the number of years beyond age 18, and the dose in any 13 consecutive weeks shall not exceed 3 rems. Thus the accumulated MPD = 5 (N-18) rems, where N is the age in years and is greater than 18.
   b. External exposure to other organs.
      Skin of whole body: MPD = 10 (N-18) rems, and the dose in any 13 consecutive weeks shall not exceed 6 rems.
      Lens of the eyes: the dose to the lens of the eyes shall be limited by the dose to the head and trunk.
      Hands, forearms, feet, and ankles: MPD = 75 rems per year, and the dose in any 13 consecutive weeks shall not exceed 25 rems.
2. Emergency dose (radiation workers)
   An accidental or emergency dose of 25 rems to the whole body or a major portion thereof, occurring only once in the lifetime of the person, need not be included in the determination of the radiation exposure status of that person.
3. Medical dose (radiation workers)
   Radiation exposures resulting from necessary medical and dental procedures need not be included in the determination of the radiation exposure status of the person concerned.
4. Dose to persons outside of controlled areas
   The radiation or radioactive material outside a controlled area, attributable to normal operations within the controlled area, shall be such that it is improbable that any individual will receive a dose of more than 0.5 rem in any year from external radiation.

2.5 RADIATION DOSIMETRY AND CONVERSION

Because of the ready availability of numerous references on radiation dosimetry, this discussion will be applied to the application of single- and multiple-collision dose conversion factors.

The common practice has been to use a single-collision response function to calibrate neutron dosimeters against Po-Be sources in air. The implied assumption that the neutrons of the incident flux can each make

at most only a single collision in the dosimeter seems logical. However, a dose measured in air cannot be equated to tissue dose because of the importance of multiple collisions and secondary reactions that would occur in thick tissue. Measured dose in water would more closely approximate tissue dose since water is nearly equivalent to tissue in its nuclear properties and provides nearly the same buildup. Point kernel calculations of doses in water should agree with dose measurements in water if the point kernels were initially fitted to experimental dose measurements in water. Using the same point kernels to compute tissue dose at positions in air would be incorrect. The computed dose in air would have to be divided by a weighted average ratio of multiple-collision to single-collision conversion factors in order to obtain a valid comparison.

Although conversion factors are readily available, single-collision conversion factors that were computed by GE-ANPD are included at this point for convenience. Factors are presented for converting gamma ray flux to absorbed dose rates in air, carbon, and tissue; neutron flux to first-collision absorbed dose rates in ethylene and tissue; and neutron flux to first-collision RBE dose rates in tissue. The conversion factors were calculated by mathematically combining known atomic cross sections, atomic abundances, and RBE factors. The results presented here consist of conversion factors for gamma ray energies ranging from 0.1 to 10 Mev, neutron conversion factors for energies ranging from 0.1 to 18 Mev, and RBE dose rate conversion factors for 0.1 to 18 Mev. Curves of these conversion factors are presented in Figures 10 to 12.

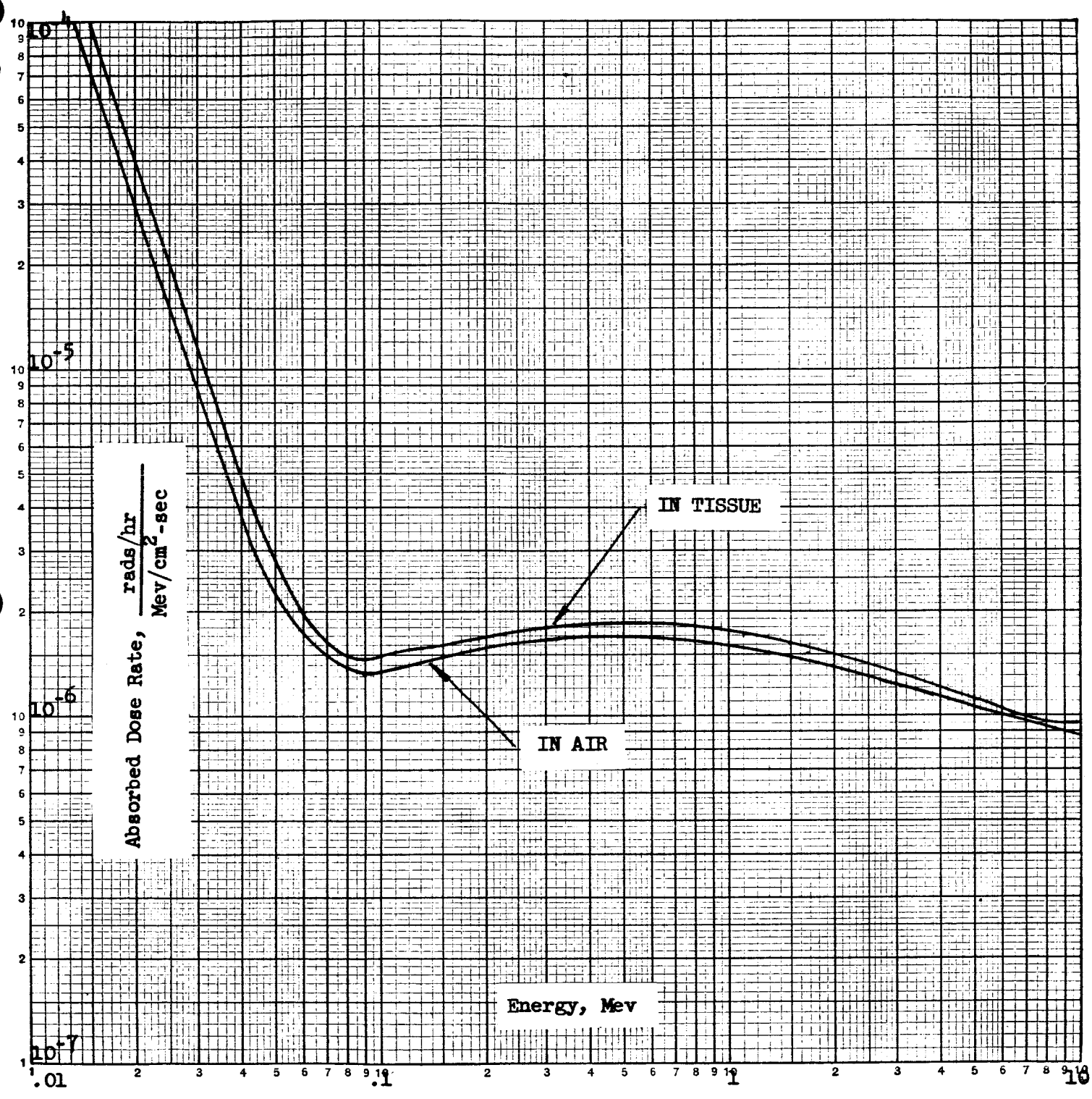


FIGURE 10

FACTORS FOR CONVERTING GAMMA RAY ENERGY FLUX
TO ABSORBED DOSE RATE

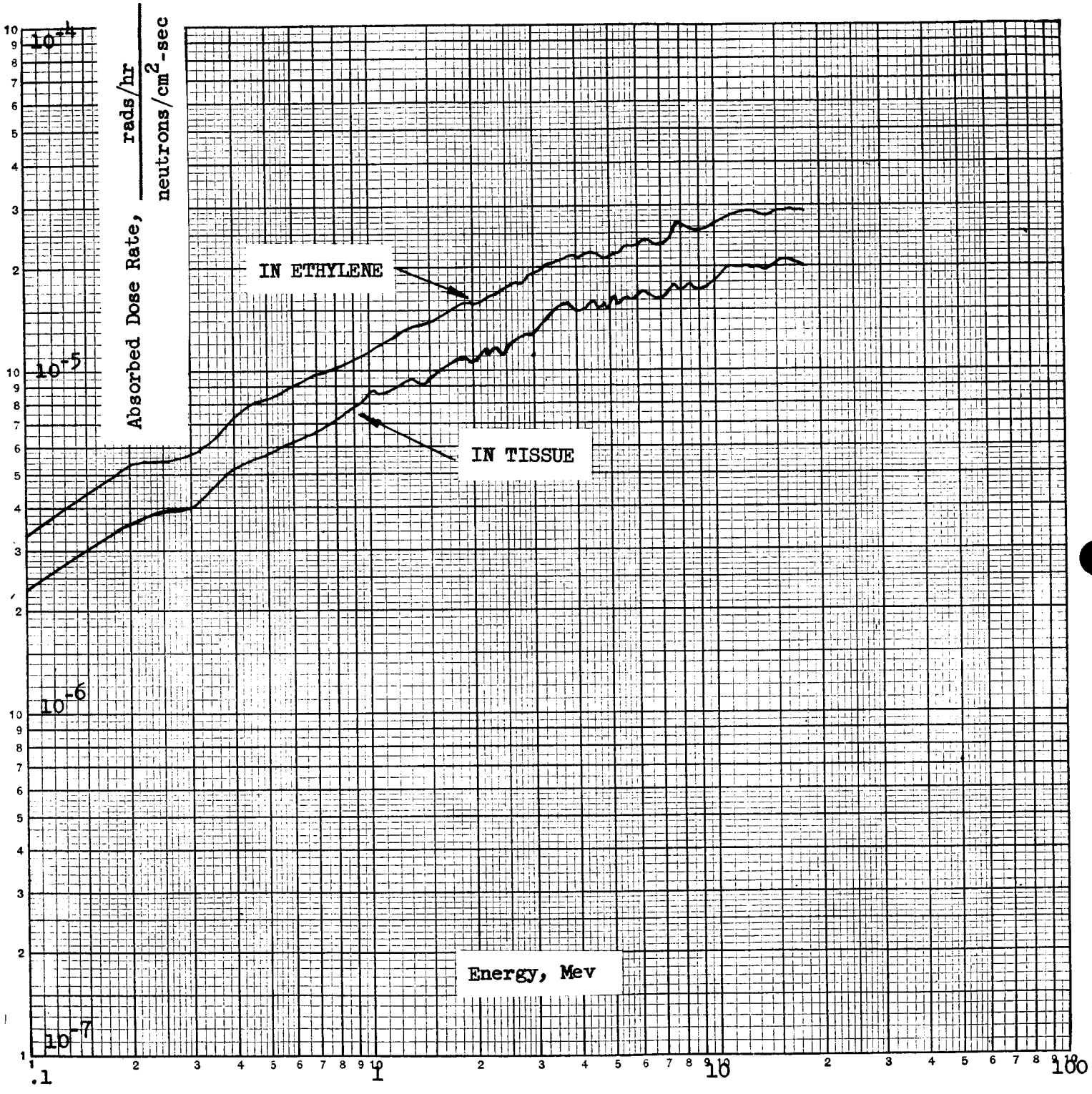


FIGURE 11

FACTORS FOR CONVERTING NEUTRON FLUX TO ABSORBED DOSE RATE

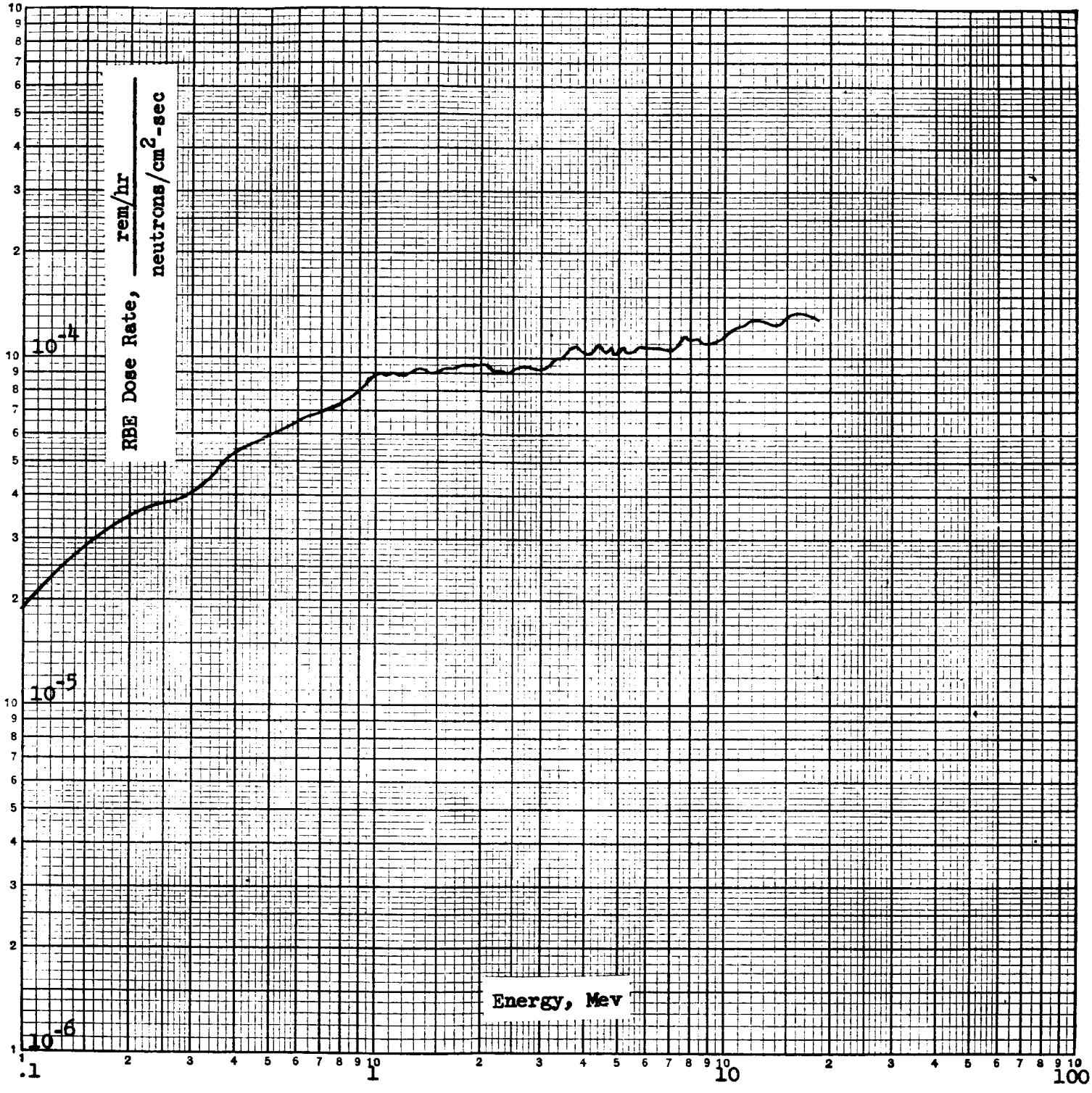


FIGURE 12

FACTORS FOR CONVERTING NEUTRON FLUX TO RBE DOSE RATE

3.0 SHIELDING CALCULATION METHODS

3.1 NARROW BEAM ATTENUATION

As a result of absorption and scattering, the particle flux intensity of a narrow monodirectional beam of neutrons or gamma photons decreases in an exponential fashion in penetrating shielding material. This can be demonstrated by assuming a narrow radiation beam of cross sectional area C cm^2 and flux intensity Φ_0 particles cm^{-2} sec^{-1} incident normally on the face of a layer of shielding material. The sketch in Figure 13 illustrates the penetration of particles through a shield. The flux intensity of the beam at a depth x cm within the material has been reduced to a value $\Phi(x)$, usually called the uncollided flux. The flux intensity per unit area of beam cross section is decreased by an additional amount $d\Phi$ in penetrating an additional distance dx. Since the cross sectional area of the beam is defined as C cm^2, the number of particles removed in the differential thickness of material is $Cd\Phi$ particles per second. The total reaction rate for particle absorption and scattering is $\Phi(x)\Sigma$ reactions cm^{-3} sec^{-1} where Σ is the total linear reaction cross section of the material. The number of particles absorbed or scattered by the volume element of cross section C cm^2 and thickness dx is, therefore, $\Phi(x)\Sigma$ Cdx reactions sec^{-1}. This is equivalent to the number of particles removed, since they are lost to the monodirectional beam by absorption or changing their direction by scattering. The following equation may be written:

Numberoof particles removed $\equiv$ number of particle reacted.

$$Cd\Phi = \Phi(x)\Sigma\ Cdx \tag{8}$$

or

$$\frac{d\Phi}{\Phi(x)} = \Sigma\, dx \tag{9}$$

on integrating, the uncollided flux $\Phi(x)$ emerging through a slab with thickness x is found to be

$$\Phi(x) = \Phi_0\, e^{-\Sigma x} \quad \text{particles } cm^{-2}\ sec^{-1} \tag{10}$$

where Φ_0 = initial flux intensity; particles cm^{-2} sec^{-1},

Σ = total linear cross section; reactions cm^{-1},

and x = shield thickness; cm at the point of interest.

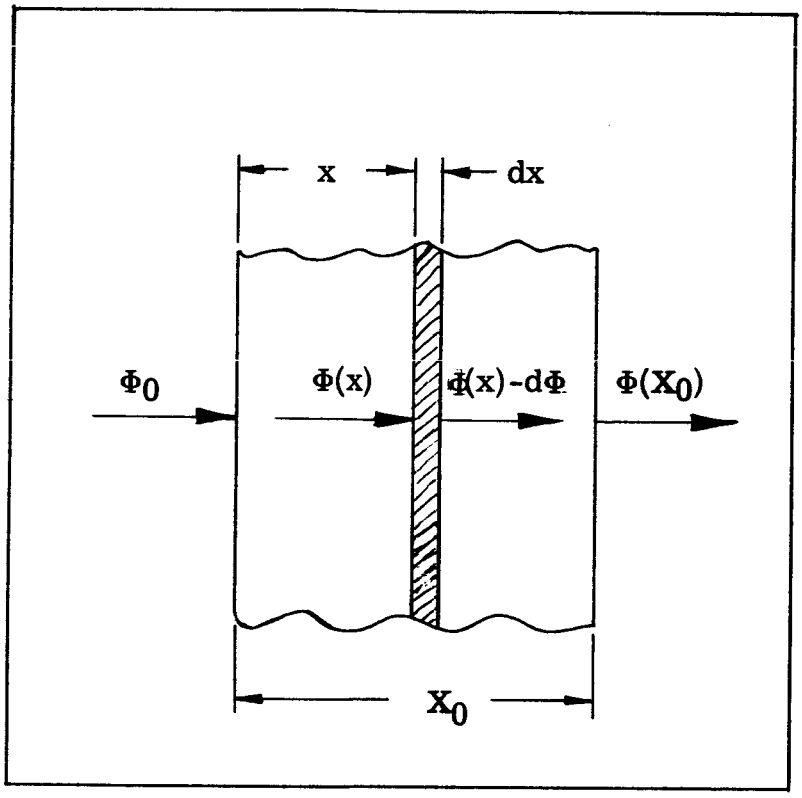


Figure 13. Particle attenuation

If the particle energy is E (Mev), the attenuation in units of energy flux becomes

$$\begin{aligned} I(x) &= \Phi(x)\ (\text{particles cm}^{-2}\ \text{sec}^{-1})\ E\ (\text{Mev particle}^{-1}) \\ &= \Phi(x)E \qquad \text{Mev cm}^{-2}\ \text{sec}^{-1} \\ &= \Phi_0 E\, e^{-\Sigma x} \qquad \text{Mev cm}^{-2}\ \text{sec}^{-1}. \end{aligned} \tag{11}$$

where I(x) is used to designate energy flux and $\Phi(x)$ is used to designate particle flux.

For two or more successive layers of different materials of thickness x_1 and x_2 whose cross sections are Σ_1 and Σ_2, Equation 10 becomes

$$I(x_1 + x_2) = \Phi_0 E\, e^{-(\Sigma_1 x_1 + \Sigma_2 x_2)} \qquad \text{Mev cm}^{-2}\ \text{sec}^{-1} \tag{12}$$

If particles of two or more different energies E_1 and E_2 are considered for a given thickness x, the following equation may be used:

$$\begin{aligned} I(x) &= I(E_1, x) + I(E_2, x) \\ &= \left[\Phi_0(E_1)E_1\, e^{-\Sigma(E_1)x} + \Phi_0(E_2)E_2\, e^{-\Sigma(E_2)x}\right] \text{Mev cm}^{-2}\ \text{sec}^{-1} \end{aligned} \tag{13}$$

In this case, the linear cross section at the specified energy must be used.

In terms of dose rate Equation 10 may be written as

$$D_{Tissue}(x) = \Phi(x)\, E\ (\text{Mev cm}^{-2}\ \text{sec}^{-1})\ K_{Tissue}\left(\frac{\text{rad(tissue)hr}^{-1}}{\text{Mev cm}^{-2}\ \text{sec}^{-1}}\right) \tag{14}$$

where K_{Tissue} is the previously defined conversion coefficient relating energy flux intensity and tissue dose rate.

3. 2 BROAD BEAM ATTENUATION WITH BUILDUP

The significance of the terms "narrow beam" in the previous section is the fact that if the radiation beam is narrow, the gamma photons are removed from the beam either by absorption or by scattering of the photons or neutrons out of the beam as it penetrates the shield. If the radiation beam is a broad monodirectional beam, it may be considered to consist of many narrow beams in parallel. In such a case, the particles scattered out of a particular narrow beam will to some extent be replaced by particles scattered in from adjacent beams. This has the effect of "Building up" the radiation level above that indicated by the

simple exponential attenuation law. For the case of photons this may be taken into account by modifying Equation 10 to:

$$I(x) = \Phi_0 E\, B_{EM}(\Sigma x)\, e^{-\Sigma x} \qquad \text{Mev cm}^{-2}\ \text{sec}^{-1} \tag{15}$$

where B_{EM} is called the monodirectional broad beam energy build up factor. In terms of dose rate Equation 15 becomes

$$D_{Tissue}(x) = \Phi_0 E\, K_{Tissue}\, B_{IM}(\Sigma x)\, e^{-\Sigma x} \qquad \text{Rads(Tissue)hr}^{-1} \tag{16}$$

The specific values of the buildup factors are presented in terms of the shield material, the initial gamma energy, and the shield thickness in relaxation lengths. The gamma energy and the dose rate conversion factors are different for the scattered and unscattered radiation. However, this effect is taken into account by the buildup factor.

If the broad beam penetrates several successive layers of shield materials, Equation 16 may be written as

$$D_{Tissue}(x) = \Phi_0 E\, K_{Tissue}\, B_{IM}(\Sigma_1 x_1 + \Sigma_2 x_2 + \dots)\, e^{-(\Sigma_1 x_1 + \Sigma_2 x_2 + \dots)}$$

$$\text{Rads(Tissue)hr}^{-1} \tag{17}$$

The value of the gamma buildup factor chosen should be a representative value corresponding to the total number of relaxation lengths $(\Sigma_1 x_1 + \Sigma_2 x_2 + \cdots)$. If all of the materials are not of the same atomic number category, the most conservative procedure is to select a buildup factor as though the entire relaxation length thickness were in the material with the highest buildup. However, it is more practical to select a value based on the assumption that the entire relaxation length thickness were in the material constituting the final layer, as long as the final layer has a thickness of one or more relaxation lengths.*

Dose buildup factors for a point Isotropic source in water, iron and lead are plotted in Figures 14, 15 and 16 respectively. For a shield thickness equivalent to 10 relaxation lengths, the dose buildup factor in water for a 1-Mev gamma photon is about 27. This may be compared to 16 for iron and 4.8 for lead for a gamma photon of the same energy. The energy absorption buildup factor for a 1-Mev photon at a shield thickness equivalent to 10 relaxation lengths of iron is 22 (see Figure 17).

*Reactor Handbook, Second Edition Vol. III, Part B, page 117

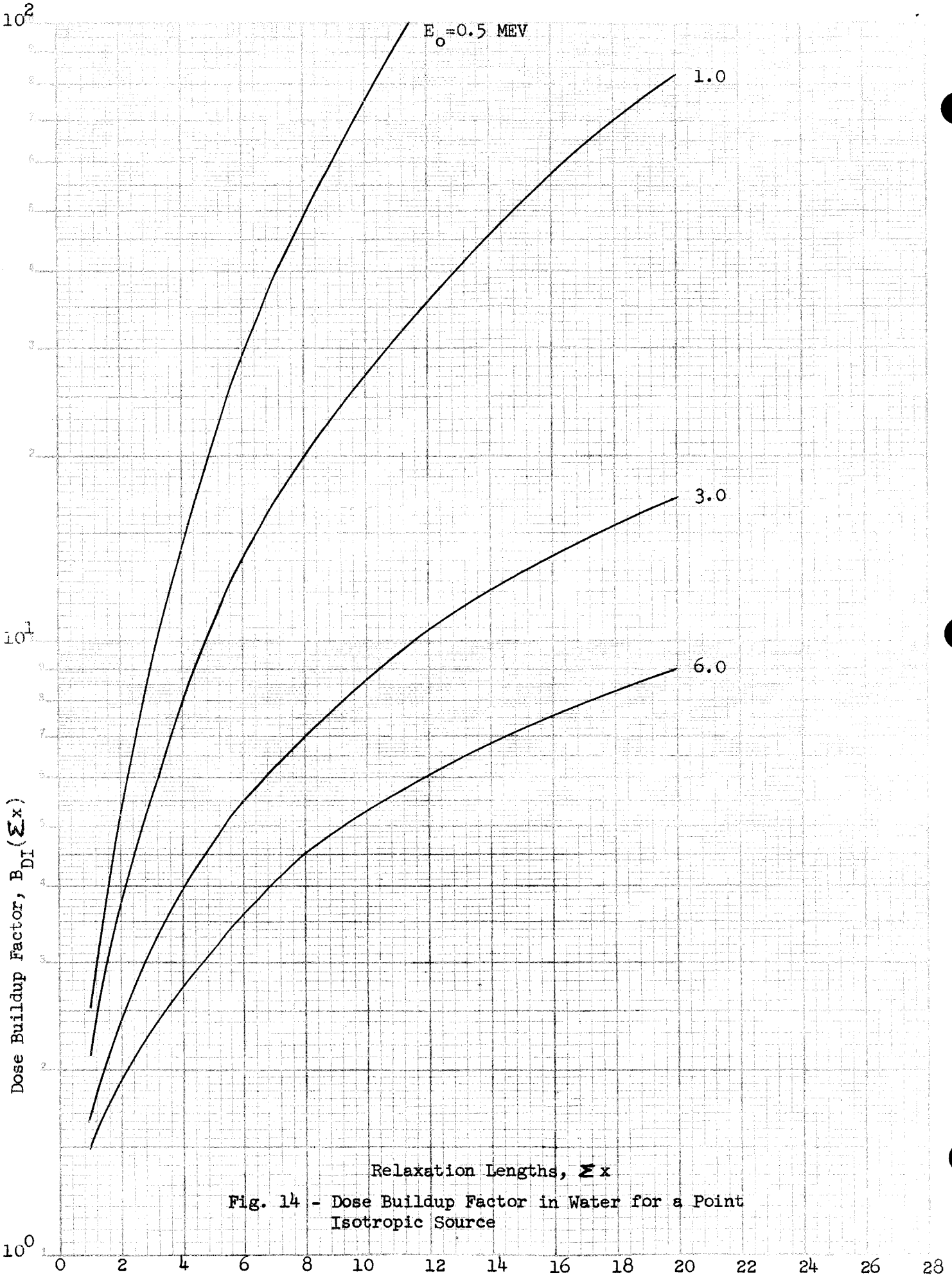


Fig. 14 - Dose Buildup Factor in Water for a Point Isotropic Source

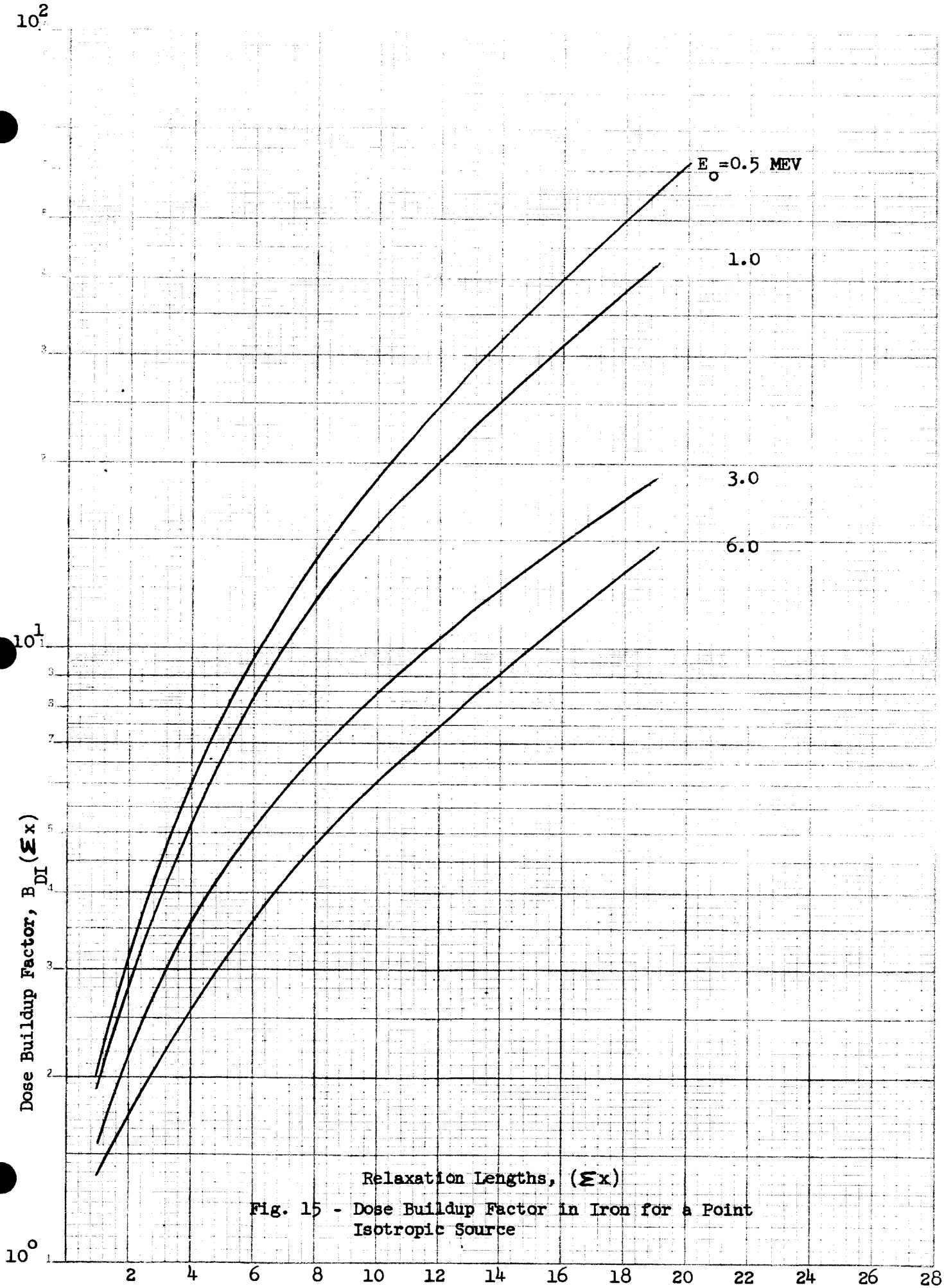


Fig. 15 - Dose Buildup Factor in Iron for a Point Isotropic Source

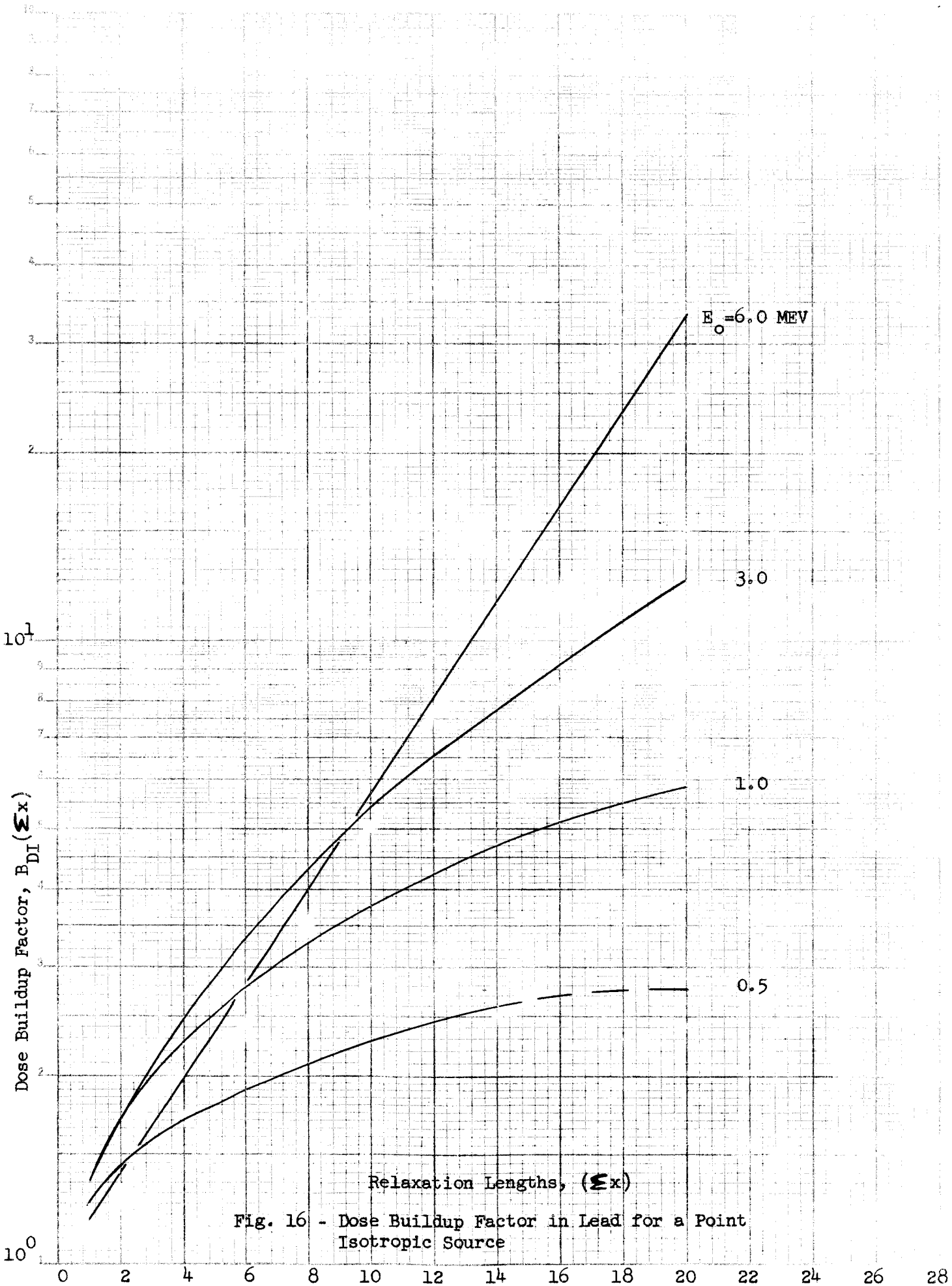


Fig. 16 - Dose Buildup Factor in Lead for a Point Isotropic Source

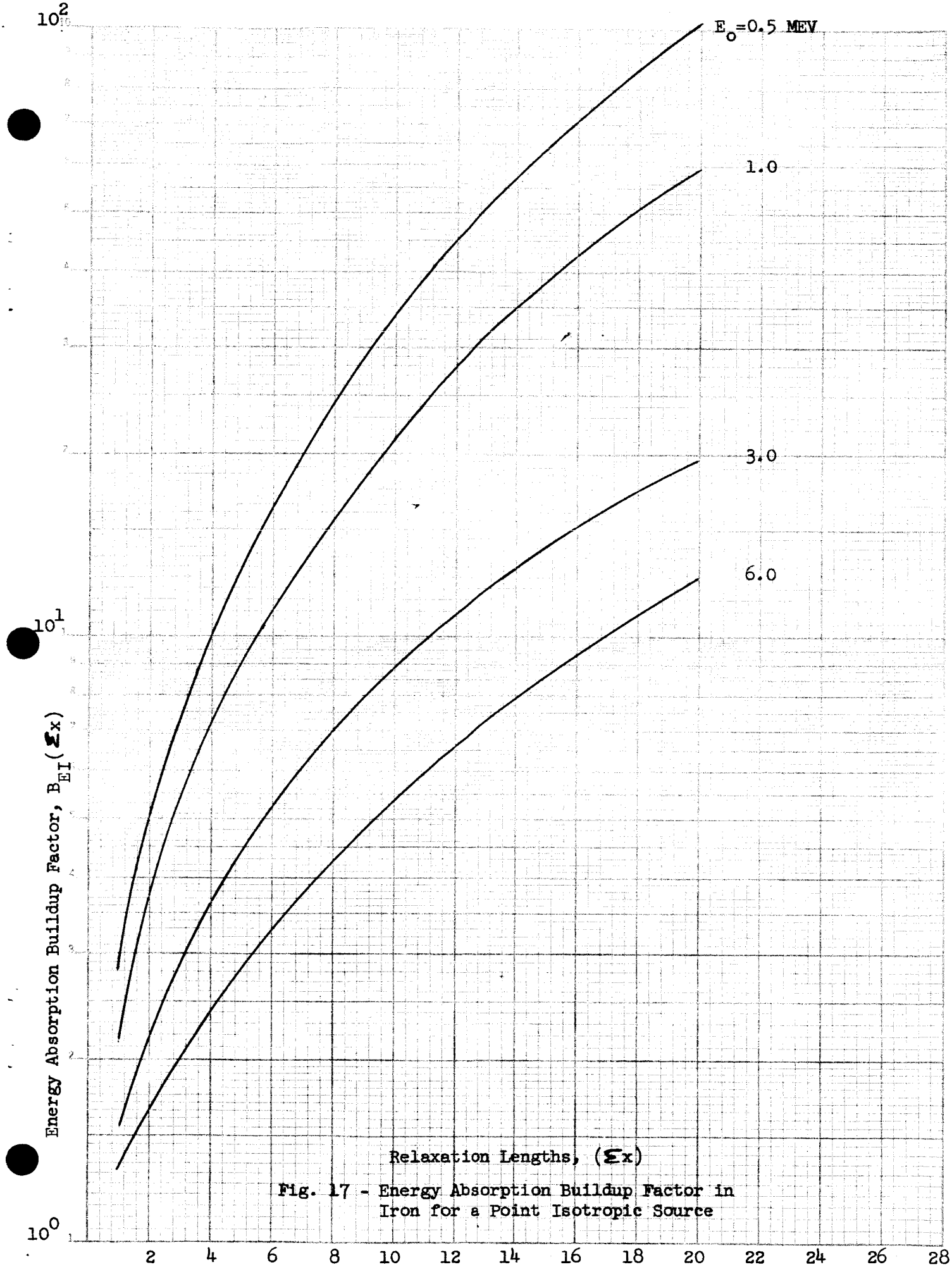


Fig. 17 - Energy Absorption Buildup Factor in Iron for a Point Isotropic Source

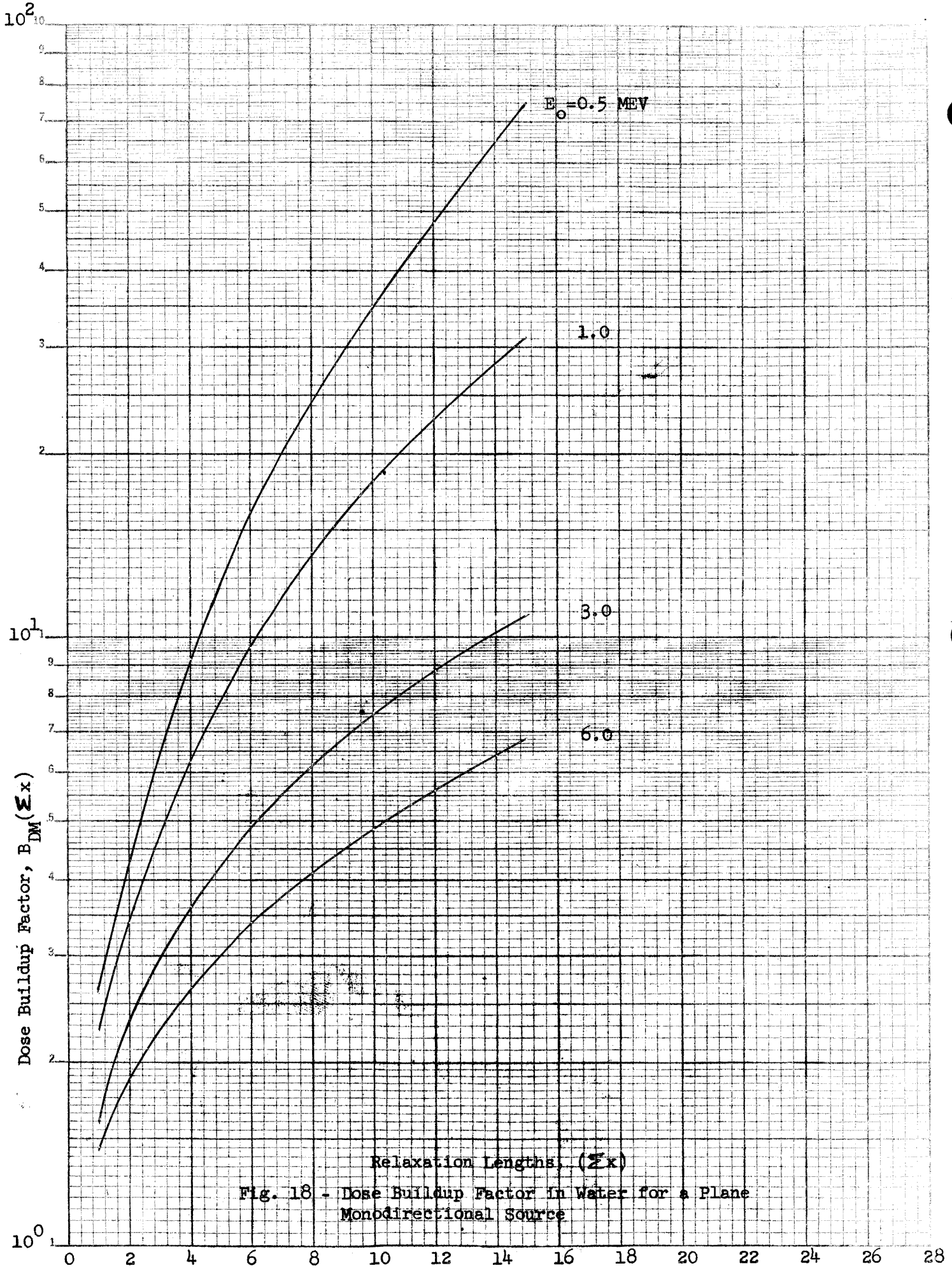


Fig. 18 - Dose Buildup Factor in Water for a Plane Monodirectional Source

The buildup or correction factor applied to the exponential attenuation term in iron for 10 relaxation lengths of shield thickness will therefore be 22 if gamma heating in iron is desired and 16 if a tissue equivalent detector (yielding dose) measurement is being compared.

Figure 18 is a plot of the dose buildup factor for water for the case of a plane monodirectional source. In this case, the buildup factor is 18 for a shield with a thickness equivalent to 10 relaxation lengths or about 2/3 the value given for the point isotropic source.

In practice, the buildup factor is usually used in connection with gamma radiation, but it could also be used in the calculation of neutron penetration. Neutron buildup factors are not now available in the more prominent literature. For neutrons a practice has been to calculate the neutron penetration to some position within the shield using the energy dependent collimated removal cross sections. The flux intensity calculated at this point is then used as the source term in a subsequent neutron diffusion theory calculation. This, in effect, accounts for the buildup of neutrons in that region of the shield. With the use of high speed electronic computers, more sophisticated methods of analysis are used.

3.3 INVERSE SQUARE LAW

3.3.1 Inverse Square Law For Isotropic Emitter

Consider the case of a radioactive point source of strength Q_p particles per second which is assumed to be emitted isotropically (same in all directions) in a vacuum. Now, assume a sphere of radius r (cm) surrounding the source. The surface area of this sphere is $4\pi r^2$ cm^2. Since all radiation must pass through the surface of the sphere, $Q_p / 4\pi r^2$ particles per second must pass through each square centimeter of surface. In other words, the flux intensity at a distance r from a point source emitting radiation isotropically is given by

$$\Phi(r) = \frac{Q_p}{4\pi r^2} \text{ particles cm}^{-2} \text{ sec}^{-1} \qquad (18)$$

This is the "inverse square law" for an unshielded isotropic source which is generally referred to as the geometrical attenuation. If the particles have an energy E, the energy flux intensity is

$$I(r) = \frac{Q_p E}{4\pi r^2} \text{ Mev cm}^{-2} \text{ sec}^{-1} \qquad (19)$$

The Dose Rate is

$$D_{Tissue}(r) = K_{Tissue} \frac{Q_p E}{4\pi r^2} \text{ rads (Tissue) } hr^{-1} \qquad (20)$$

3. 3. 2 Inverse Square Law With Attenuation and Buildup

The inverse square law for radiation from an isotropic point source expressed in terms of Dose Rate may be modified by exponential attenuation and buildup to be

$$D(x, r)_{Tissue} = K_{Tissue} B(\Sigma x)_{DI} \frac{Q_p E}{4\pi r^2} e^{-\Sigma x} \text{ rad(tissue)} hr^{-1} \qquad (21)$$

where B_{DI} is the point isotropic source dose rate buildup factor. This differs somewhat in value from the broad beam monodirectional dose rate buildup factor B_{DM}. When shield layers of widely different atomic numbers are used, the buildup factor should be handled in the same manner as described for the broad beam buildup factor.

If the source emits particles of several different energies, it is necessary to make separate calculations for the number of particles emitted of each energy and to add the separate results to obtain the total dose rate.

3. 4 RADIATION FROM SHIELDED DISTRIBUTED SOURCES WITH SELF-ABSORPTION

3. 4. 1 General Method

A reactor or any radioactive object may be assumed to consist of small volume elements, each of which is treated as a point source. The total flux intensity at any external position is then the sum of the separate contributions.

Each individual point source is shielded not only by the shield but by the remainder of the reactor. This shielding effect of the source is known as "self-shielding" or "self-absorption." In calculating the attenuation from each point source, it will be found that each is shielded by a different thickness of shield. To most of these point sources the shield will look thicker than its nominal thickness due to the penetration of radiation at slant angles through the reactor and through the shield. This has an important effect on reducing the resultant radiation level at locations immediately outside of the shield if the radioactive object is distributed

over an extended region within the shield. (A long radioactive fuel element closely encased within a lead cask is a particularly important case in which the surface radiation level is significantly reduced by slant penetration through the shield of radiation from the ends of the fuel element.) By drawing a line from each of the assumed point sources to the position of the receiver one may measure the thicknesses x_1, x_2, x_3, $\cdots$ of reactor and shielding materials which must be traversed and the distance r to the receiver. Equation 21 may then be used to calculate the dose rate from each source point. If the source emits particles of different energies, the calculation must be repeated for each energy group of particles and must also be repeated for each of the assumed point sources and the results added to obtain the total dose rate. The integration procedure may be performed either analytically or numerically.

Equations and tables for analytically calculating the integrated flux intensity and dose rate for a number of sources and shields of uniform compositions and with simple geometries are available in the listed references. Real source and shields are usually not uniform in composition and usually have complex geometries containing structure, voids and irregularities in shape. These can frequently be approximated to sufficient accuracy by the simple geometries such as spheres, cylinders, slabs and lines treated in the references.

For more complex geometries a numerical integration can be made. In the ANP Program, aircraft shield calculations were usually made by assuming a large number of point sources, measuring the shield path from each and calculating and numerically adding the separate dose rate contributions with the use of electronic computers.

3.4.2 Point Source Approximation

In many cases, the distance separating the operator or some receiver from the radioactive source is usually large compared to the dimensions of the source. In such cases approximations can be made relative to distance, self absorption and shield thickness which simplify the dose rate calculations.

Distance Approximation - When the relative distance r between the receiver position and various points within the radioactive object is large, a single representative value of the distance r can be chose. To be conservative this distance can be measured from the surface of the object to the receiver position. In practice a position one fourth of the way from the surface to the center of the object can be considered to be the effective location of the point source. The representative value of r should be measured from this effective point source location to the receiver position.

Self-Absorption Approximation - A further approximation can be made in determining self-absorption of radiation within the radioactive object since when it is viewed from a distant position the slant range attenuation within the source is not important. Relatively simple analytical expressions can then be derived for the fractional amount of radiation escaping the sides or ends of simple cylinders or spheres with a uniform concentration of radioactive material. A leakage fraction may be calculated which can be applied to the entire object without the necessity of breaking it into smaller elements.

As a consequence of the distance and self-absorption approximations, the radiation level from unshielded radioactive sources can be calculated by assuming the entire source to be a point source with a leakage fraction F, located at a representative distance r from the receiver position. The flux intensity at a distance r from an unshielded isotropic source with self-absorption then becomes

$$\Phi(r) = \frac{QF}{4\pi r^2} \text{ particles cm}^{-2} \text{ sec}^{-1} \qquad (22)$$

and the dose rate becomes

$$D(r)_{Tissue} = K_{Tissue} \frac{QEF}{4\pi r^2} \text{ rads(Tissue)hr}^{-1} \qquad (23)$$

Q is the source strength of the entire reactor or volume source in particles per second and QE the source strength in Mev per second and is assumed to be concentrated at a point at the center of the source.

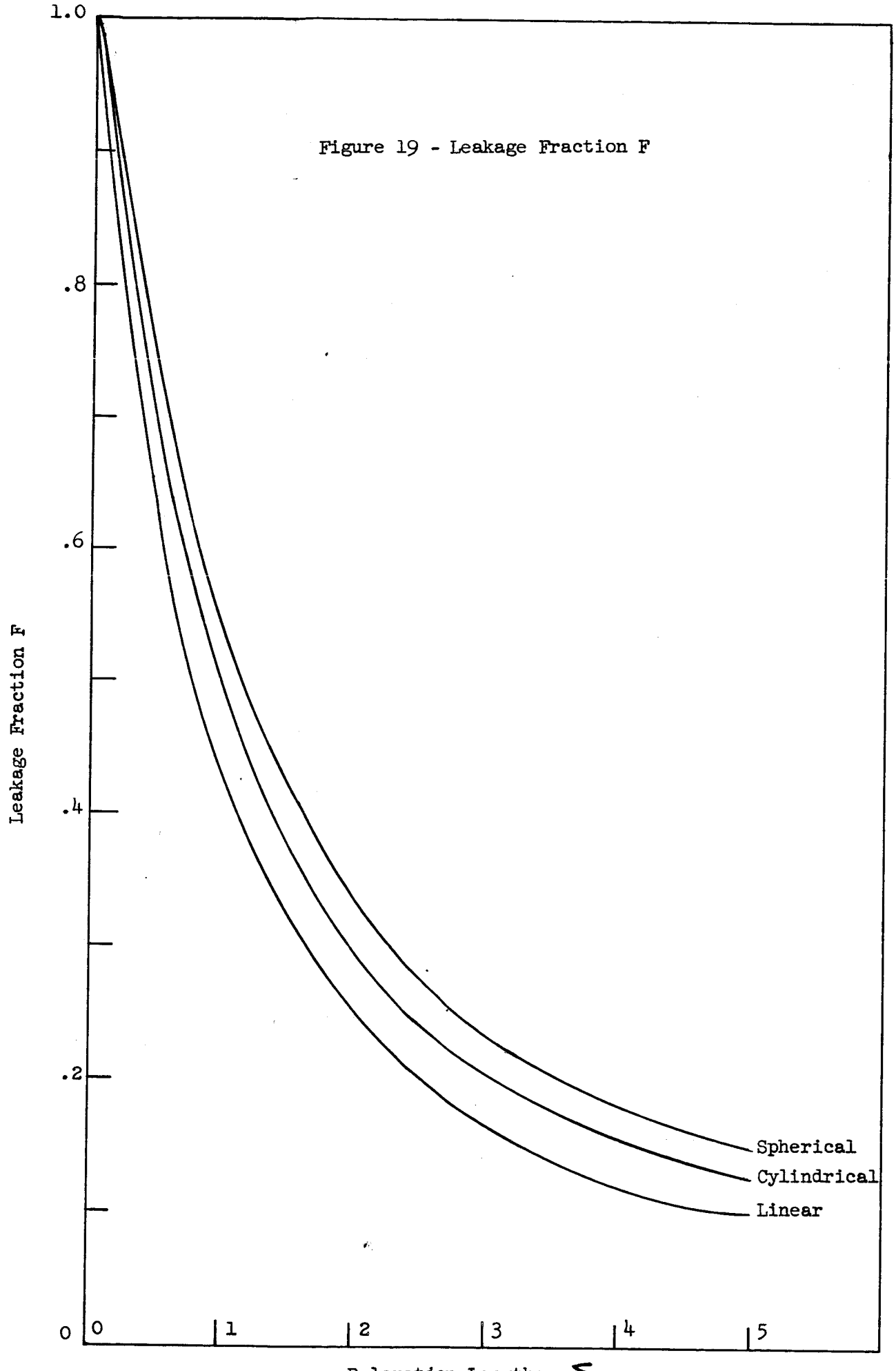
The leakage fraction F in Equations (22) and (23) is a function of the cross section Σ of the source material and the radius a of a cylindrical or spherical source or the half length a of a linear source and is plotted in Figure 19. For values of Σa greater than 5, the source leakage fraction F may be approximated by

$$F = \frac{1}{2\Sigma a} \text{ for a cylindrical source viewed from the end} \qquad (24A)$$

$$F = \frac{2}{\pi\Sigma a} \text{ for a cylindrical source viewed from the side} \qquad (24B)$$

$$F = \frac{3}{4\Sigma a} \text{ for a spherical source} \qquad (24C)$$

Figure 19 - Leakage Fraction F



The self-absorption factors given in Figure 19 and Equations (24A), (24B) and (24C) do not include provision for buildup of scattered radiation within the source. For the case of gamma radiation, the buildup within the source may be taken into account with sufficient accuracy by using the energy absorption coefficient Σ_E instead of the total cross section Σ in calculating the value of Σa. This will slightly underestimate the self-absorption, thus slightly overestimating the radiation leakage.

Shield Thickness Approximation - Another approximation may be made relative to shield thickness. When viewed from a distant position, the differences in shield path length from various points in the source will be small. It is reasonable to assume that all parts of the source are shielded by the nominal thickness of the shield.

Combining the thickness approximation with the self-absorbing point source approximation, a general point source approximation equation for the dose rate due to a shielded source with self-absorption may be written as

$$D(x, r)_{Tissue} = K_{Tissue} B_{DI}(\Sigma x) \frac{QEF}{4\pi r^2} e^{-\Sigma x} \text{ rads(Tissue)hr}^{-1} \quad (25)$$

where x is the nominal thickness of the shield measured along a straight line between the center of the source and the position of the receiver. The buildup factor $B_{DI}(\Sigma x)$ corresponds only to the nominal shield thickness (Σx) in relaxation lengths. The leakage fraction F contains both the attenuation and buildup in the source when used as previously recommended.

Using Equation (23), the above equation (Equation 25) in terms of the unshielded source, may be written as

$$D(x, r)_{Tissue} = D(r)_{Tissue} B_{DI}(\Sigma x) e^{-\Sigma x} \text{ rads(Tissue)hr}^{-1} \quad (26)$$

If the dose rate is to be calculated for several different shield thicknesses, it is usually advisable to calculate $D(r)_{Tissue}$ (unshielded) first using Equation (23) and then use Equation (26).

These approximate methods are sufficiently accurate for the condition of large source-receiver distances. From a practical viewpoint, the approximate method can also be used for positions close to the shielded source since the error will be such as to overestimate the radiation level. Even for positions directly on the surface of the shield with the source directly inside the shield, it is improbable that the radiation level in

practical cases will be overestimated by more than a factor of 2 to 5 as compared with the more accurate methods. This is acceptable in most cases since little time is usually spent in such close proximity by operating personnel. However, for special cases where prolonged personnel or equipment exposure is planned for operations very close to the shield surface with sources of relatively large dimensions compared to the shield thickness, it is advisable to make the more accurate calculation to avoid over-conservatism in radiation exposure or in shield thickness and weight.

3.4.3 Surface Current Method

If the surface radiation current is already known for a radioactive object, it may be treated as an assembly of surface source elements, rather than volume elements.

For example, assume that the current through the surface of a reactor is known to be J gammas cm^{-2} sec^{-1}. A surface element of area ΔS cm^2 acts like a point source whose strength is $Q_p = J\Delta S$ gammas/sec^{-1}. The particles which leave such a point source may proceed in any direction into the region outside the surface, but the emission per unit solid angle is greatest in the direction predominately normal to the surface. Consequently, the source strength will appear different to observers at various angles θ to the surface normal. The flux intensity from such a surface element at a distance r is closely approximated by the expression

$$\Phi(r, \theta) = \frac{2J\Delta S \cos\theta}{2\pi r^2} \text{ particles cm}^{-2} \text{ sec}^{-1} \qquad (27)$$

Presumably self-absorption in the reactor has already been considered in determining the value of the surface current J. Hence, no separate source leakage factor is needed. More careful consideration of a specific internal source distribution, and self-absorption may result in a different angular dependence such as $\cos^2\theta$ or combinations of isotropic and cosine distributions. However, for first approximations, the assumption of a simple cosine θ distribution for the case described is adequate.

It should be noted that the cosine distribution applies to cases where the real sources are internal and the surface element of source strength $J\Delta S$ particles per sec represents a fictional, although convenient point source. If on the other hand there are no internal sources but all the radioactive material is deposited on the surface so that Q_S particles per second are being emitted per cm^2 of surface, then each surface element should be treated as an isotropic emitter of source strength $Q_p = Q_S\Delta S$ particles per sec^{-1} for use in Equation (25).

Rules regarding the choice of the distribution law for radiation leaving the surface of a contained radioactive source may be listed as follows.

Spherical Distribution - When self-absorption in the source is unimportant and the source is homogeneous (i. e., the activity per unit volume is constant throughout the source), it may be considered that the rays from that part of the source which is "looked at" by an observer are uniformly distributed in all directions. In other words, a fraction $1/2\pi$ of the particles emitted through 1 cm^2 of surface go into unit solid angle.

Cosine Distribution - A self-absorbing, homogeneous source such as a slab of active metal or a body of active solution emits more strongly in the forward direction, i. e., along the normal surface of the slab. If the flux through the surface is J, the number emitted into unit solid angle at the angle θ is $(2J \cos\theta)/2\pi$. (See Equation 27.)

Fermi Distribution - A self-absorbing, inhomogeneous source such as a reactor, in which the activity per unit volume increases with increasing distance below the surface, emits even more strongly in the forward direction. A reasonable approximation for this situation is the Fermi distribution, according to which the number into unit solid angle at the angle θ is $[0.93\ J(\cos\theta + \sqrt{3}\cos^2\theta)]/2\pi$. In this case Equation (27) would become

$$\Phi(r, \theta) = \frac{0.93 J \Delta S [\cos\theta + \sqrt{3}\cos^2\theta]}{2\pi r^2} \text{ particles cm}^{-2} \text{ sec}^{-1} \qquad (28)$$

3. 4. 4 Two-Component Method

In the case of a shielded source of neutrons, for instance, the radiation to some receiver point may be assumed to arrive from two distinct regions of the reactor-shield assembly. The first component, which is obtained by a volume integration over the active core of the reactor, consists of neutrons that are essentially uncollided and/or neutrons that have undergone very low angle elastic scattering. This narrow beam of radiation is called the collimated component.

The second component which is obtained by a surface integration over the major shield exterior surface consists of neutrons that have undergone large angle elastic scattering and neutrons that have been inelastically scattered and reduced in energy. This radiation which has scattered several times is called the diffuse component.

Consider the case of a point source located in the finite shield as shown in Figure 20. The radiation calculated to some detector point will consist

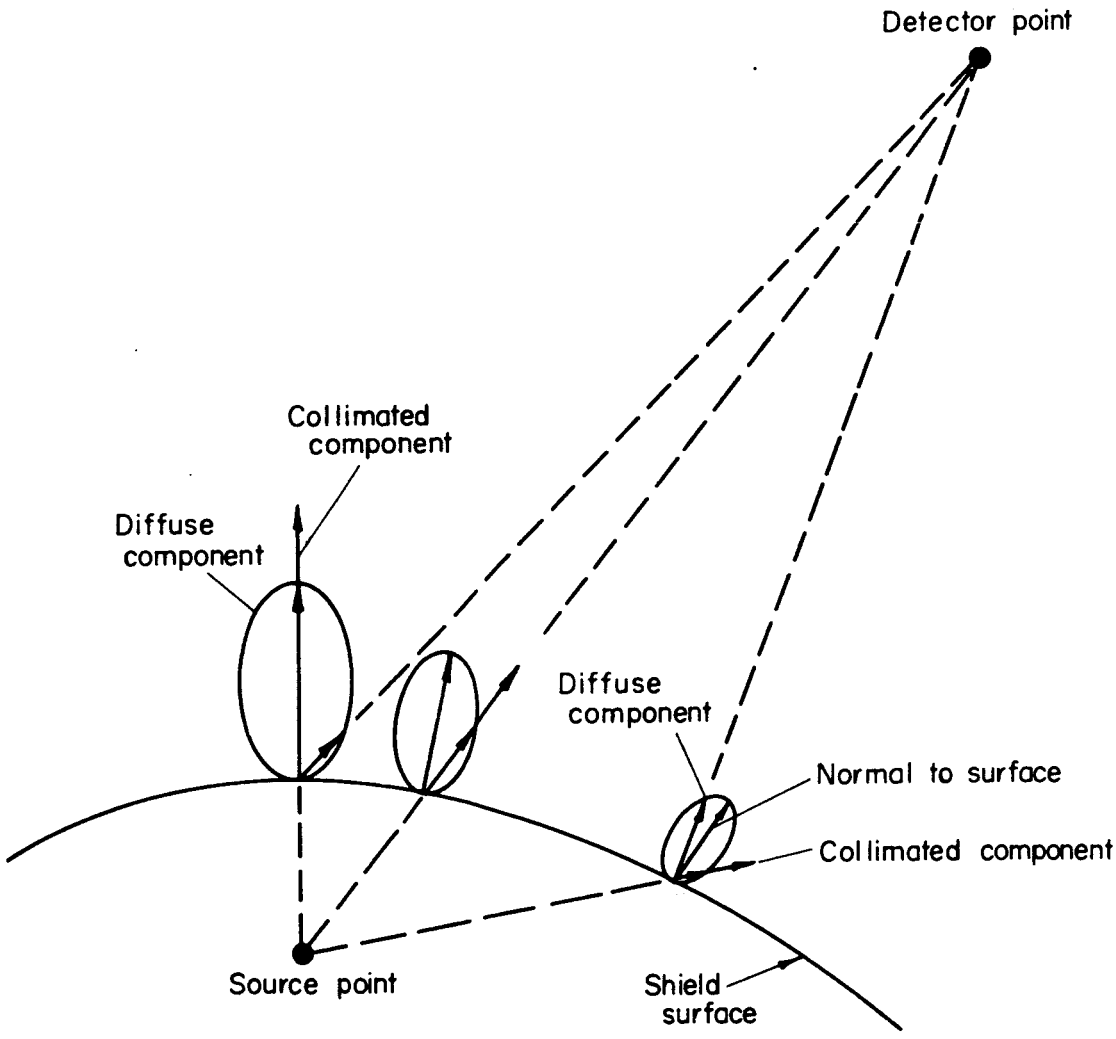


Figure 20 - Sketch Illustrating the Two-Component Concept

of a contribution by the collimated component, which considers only the linear path connecting the source and the detector, plus a usually larger contribution by the diffuse component from the surface. The difference in magnitude between the two components may be attributed primarily to the fact that (a) the isotropic detector subtends a large solid angle of the surface source compared to the point source, (b) the collimated component may at certain angles see local thicker-than-average portions of the shield along the linear path, and (c) the cross sections used for each component are not the same.

A knowledge of both the collimated and the diffuse radiation leaving the reactor shield assembly is required for the optimum design of a shield, especially for the case of a shadow shield considered in space applications.

3. 5 HEAT GENERATION IN SHIELDS

The absorption of neutrons and gamma rays in reactor shields releases radiation energy which is ultimately converted to thermal energy. This heat release is mostly concentrated on the inner sides of the shield, where special cooling may become necessary. As a general rule heat production in the shield may come from three primary sources. These sources are (a) the gamma radiation from the reactor, (b) the fast neutron radiation from the reactor, and (c) the gamma radiation or the charged particles which are emitted from the nuclides when neutrons are captured within the shield material. Although the thermal neutrons incident to the inner part of the shield are not important throughout the shield, they may contribute appreciably to the energy release at the shield-reflector interface.

Nuclear heat generation plays a major consideration in the shield design, especially when the shield system is designed for a compact reactor operating at high power, a situation in which large temperature increases may be expected. The allowable temperature increases are determined principally on the basis of thermal stress. Materials with low thermal conductivity may experience a very appreciable temperature rise; and consequently, the thermal stresses from the temperature differences within a shield may cause cracking or deformation which effect the integrity and performance of the shield. This effect may sometimes be avoided by dividing the shield into a number of smaller regions which are independently supported or by placing a thermal shield between the reactor and the bulk shield, the latter often referred to as the biological shield. The purpose of the thermal shield is to absorb much of the low energy neutrons and gamma radiation in a material that may be easily cooled so that the rest of the bulk shield will not receive a large thermal (or heat generation) gradient.

3. 5. 1 Neutron Heating

Energy absorption from neutron energy degradation by shield materials may be calculated at any point in the shield from the relation

$$H_n(x) = C_0 \int \overline{\Delta E}(E, x)\, \Sigma_s(E, x)\, \Phi(E, x)\, dE \quad \text{watts cm}^{-3} \tag{29}$$

where $H_n(x)$ = heat generation at x;

$\overline{\Delta E}(E, x)$ = average energy loss per scattering collision for neutrons of energy E at x;

$\Sigma_s(E, x)$ = macroscopic scattering cross section at x for neutron of energy E;

$\Phi(E, x)$ = flux of neutrons with energy E at x;

C_0 = 1. 6 x 10^{-13} watt sec Mev^{-1}.

The integration extends over all energies present.

3. 5. 2 Charged-Particle Heating

Energy absorption of charged particles which are emitted in the capture of thermal neutrons such as the boron, (n, α) reaction, is important principally for a few slight elements and for fissionable materials that may be in the shield. Energy absorption from charged-particle emission is given by

$$H_C(x) = C_0 \int E_C\, \Sigma_{aC}(E, x)\, \Phi(E, x)\, dE \quad \text{watts cm}^{-3} \tag{30}$$

where $H_C(x)$ = heat generation at x;

$\Sigma_{aC}(E, x)$ = macroscopic neutron cross section for charged-particle emission at x;

E_C = charged-particle energy;

$\Phi(E, x)$ = neutron flux at x for neutrons with energy E;

C_0 = 1. 6 x 10^{-13} watt sec Mev^{-1}.

3. 5. 3 Gamma Heating

Energy absorption of gamma rays by shield materials may be calculated at any point in the shield from

$$H_\gamma(x) = C_0 \int E\, \Sigma_{aE}(E, x)\, \Phi_\gamma(E, x)\, dE \tag{31}$$

where $H_\gamma(x)$ = heat generation at x;
E = energy of gamma rays;
$\Sigma_{aE}(E, x)$ = energy absorption coefficient of the shield materials for gamma rays of energy E at x;
$\Phi(E, x)$ = flux of gamma rays with energy E at x;
C_0 = 1.6×10^{-13} watt sec Mev

Capture gamma heating requires the knowledge of the neutron flux throughout the shield. Based on this neutron pattern, the number of neutron captures or the number of neutrons which are inelastically scattered are then determined. This information will then yield the number of gamma rays and the respective energies which are generated. By assuming that these gamma rays are emitted isotropically in the shield, a calculation of the gamma heating may be made. Therefore, in addition to the gamma rays that originate within the reactor the value of $\Phi_\gamma(E, x)$ in Equation (31) should also include the gamma rays which are generated in the shield. Due to the complex nature of the calculation of this form of shield heating, it is recommended that reference be made to a more detailed presentation. As a rule, however, a Monte Carlo computer program is used to solve this type of heating since the part of the shield that may be critical (from the standpoint of overheating and the subsequent thermal stress limitations) is irregular in shape and, therefore, not easily simulated for a rigorous analytical solution.

3.5.4 Nuclear Heating in Liquid Hydrogen

In the development of nuclear powered rockets, an accurate determination of the nuclear radiation heating in the rocket fuel and of the radiation transmitted to the payload region is required. Since liquid hydrogen is a likely fuel for propulsion, the accurate calculation of radiation heat deposition and radiation transfer in this media is of considerable importance for obtaining an optimum design.

A set of calculations* obtained by the use of Monte Carlo techniques for the case of plane slab geometries and right circular cylinders with flat ends have been made for a family of monoenergetic neutrons and gamma rays. A curve showing the relative heating rates of 3-Mev neutrons and gamma rays as a function of the depth in a liquid hydrogen slab is given in Figure 21. The calculations were based on 1 particle cm^{-2} sec^{-1} incident normal to the slab surface.

*MTP-RP-62-1, M. O. Burrell, "Nuclear Radiation Transfer and Heat Digestion Rates in Liquid Hydrogen," February 1962.

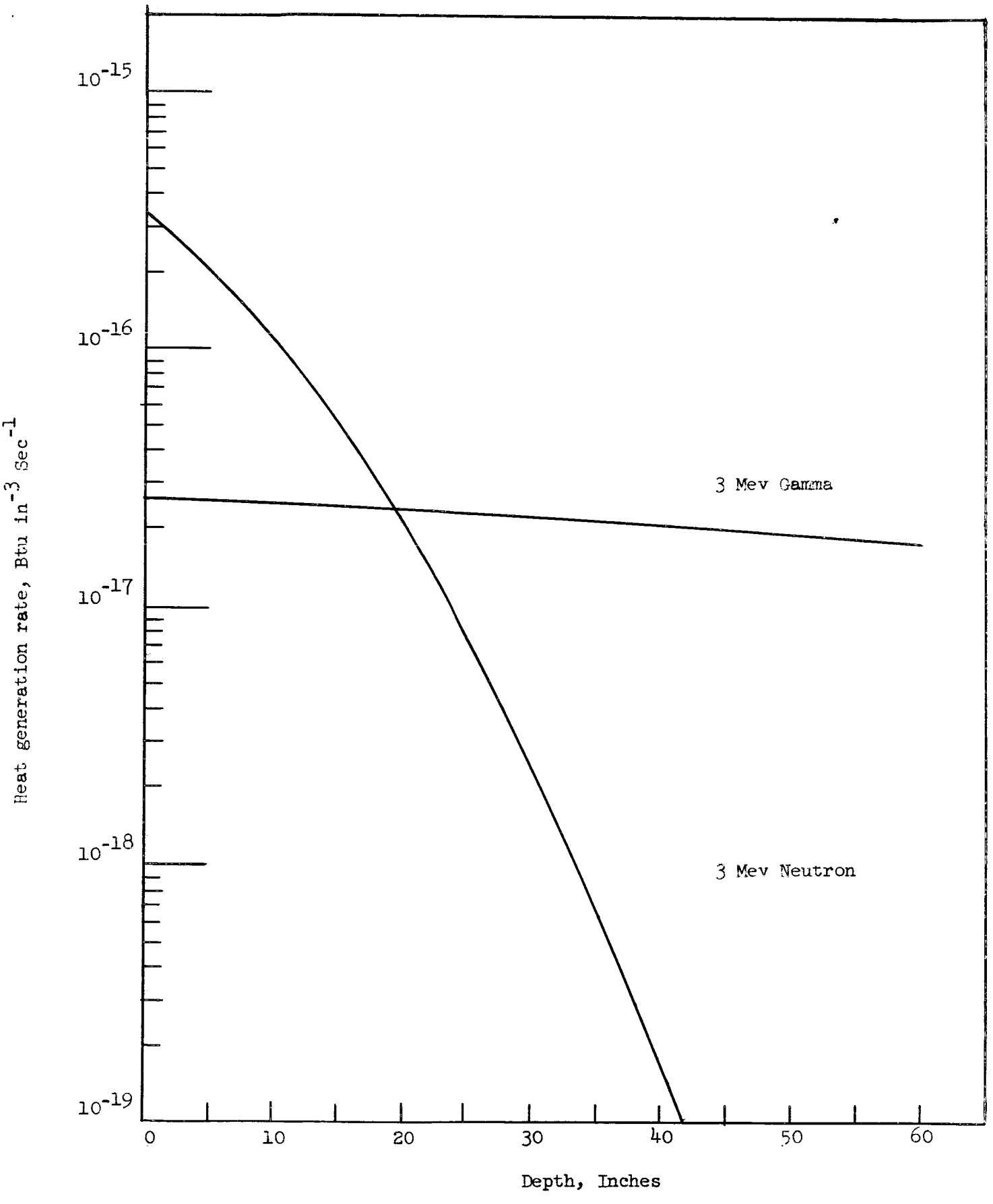


Figure 21 - Neutron and Gamma Heating Rate in Liquid Hydrogen

3. 6 SCATTERING OF RADIATION

3. 6. 1 Gamma Scattering

The scattering of radiation plays an important role in the design of divided shield propulsion systems and also in the design of power plant test facilities.

The calculation of the scattered radiation may be made in the following steps:

a. Break the scatterer into a number of mass or volume elements
b. Determine the flux intensity and energy of radiation at the position of each of these elements
c. Assume that each element is a point source of the scattered radiation and calculate the flux intensity and energy of the radiation scattered to the observer
d. Add the separate contributions from each scattering element to obtain the total flux intensity or dose rate at the observer position

The gamma ray of initial energy E_1 is then reduced in energy in the scattering process to a value, E_2, where the ratio E_2/E_1 is given by

$$P \equiv \frac{E_2}{E_1} = \frac{1}{1 + \frac{E_1}{0.51}(1 - \cos\theta)} \tag{32}$$

This ratio is given in Figure 22 for various energies E_1 and the scattering angle θ.

3. 6. 1. 1 Thin Scatterers

For thin scatterers such as viewing mirrors, thin shells and structural materials, the absorption of the gamma rays may be neglected. In this case the scattering medium (element) with a thickness Δx (cm) and area ΔS (cm^2) with a density ρ (gm cm^{-3}) will have a mass

$$\Delta M = \rho \, \Delta S \, \Delta x \quad \text{gms} \tag{33}$$

The gamma ray flux at a distance r_2 due to scattering through an angle θ from an individual element of mass ΔM is

$$\Phi_2(r_2) = 6.023 \times 10^{23} \, \Delta M \left[\frac{Z}{A}\right] \Phi_1(r_1) \, \frac{1}{r_2^2} \, \frac{d\sigma_{es}}{d\Omega} \quad \text{gammas cm}^{-2} \, \text{sec}^{-1} \tag{34}$$

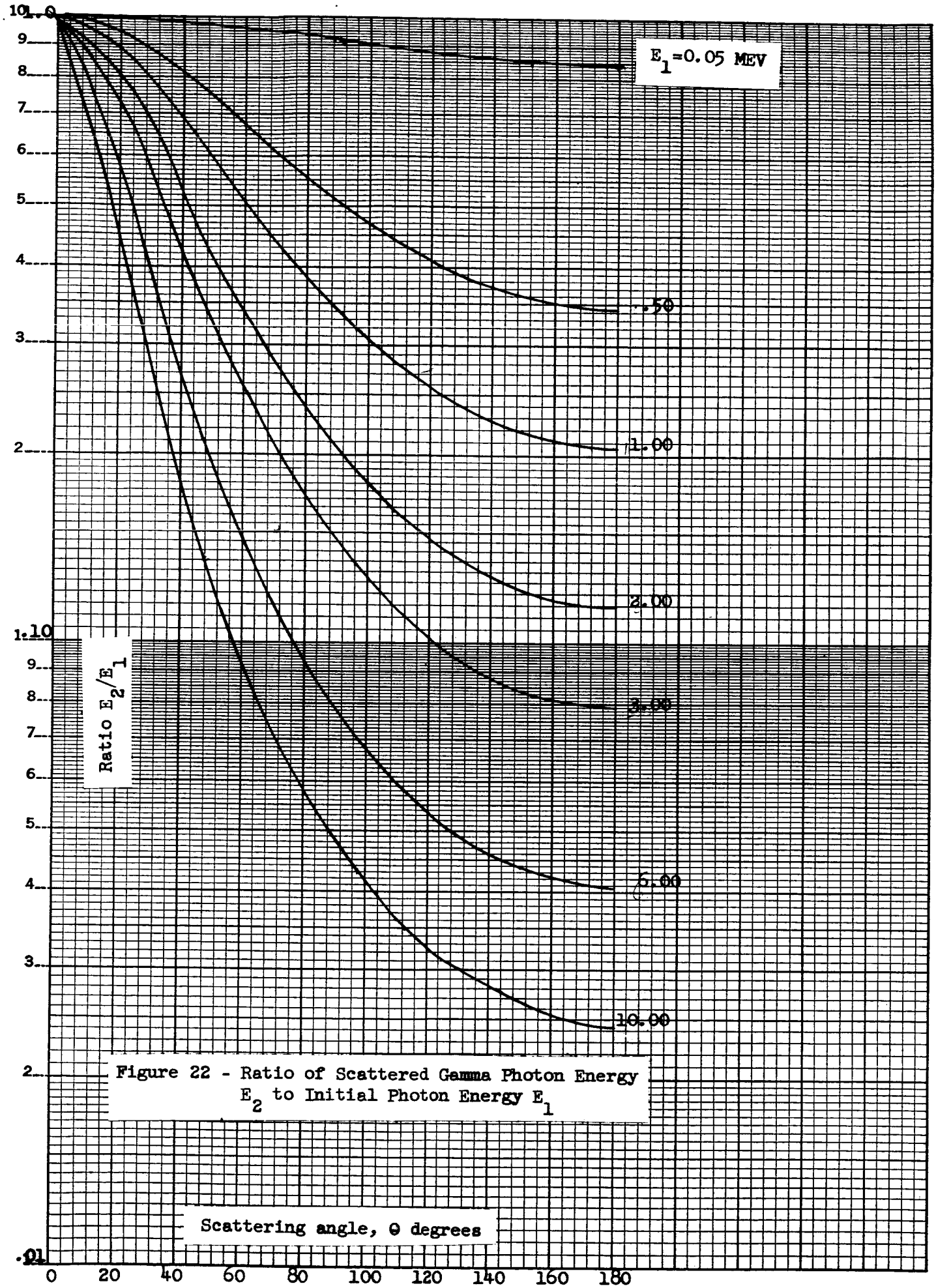


Figure 22 - Ratio of Scattered Gamma Photon Energy E_2 to Initial Photon Energy E_1

where 6.023×10^{23} is Avagadros' number and

Z = atomic number of scattering material

A = atomic weight of scattering material

$\frac{d\sigma_{es}}{d\Omega}$ = Klein-Nishina differential scattering cross section corresponding to the energy E_1 of the incident gamma photon

$\Phi_1(r_1)$ = incident gamma ray flux intensity, gammas cm^{-2} sec^{-1}

The Klein-Nishina differential scattering cross section gives the number of gamma photons scattered by a single free electron into a unit solid angle at an angle θ. It is related to the scattering angle and the gamma energies by the equation

$$\frac{d\sigma_{es}}{d\Omega} = \frac{1}{2}\left(\frac{e^2}{mc^2}\right)^2 \left[\frac{E_2}{E_1} - \left(\frac{E_2}{E_1}\right)^2 \sin^2\theta + \left(\frac{E_2}{E_1}\right)^3\right] \qquad (35)$$

where $\left(\frac{e^2}{mc^2}\right)$ is the classical radius of the electron.

The incident gamma ray flux intensity $\Phi_1(r_1)$ is given by Equation (18)

$$\Phi_1(r_1) = \frac{Q_p}{4\pi r_1^2} \text{ gammas } cm^{-2} \text{ } sec^{-1}$$

The number of electrons in M grams of an element of atomic weight A and atomic number Z is

$$M\left[\frac{Z}{A}\right] \quad \left[6.02 \times 10^{23}\right] \quad \text{electrons}$$

For most elements Z/A is about 1/2. Hence, the scattered gamma photon flux at a distance r_2 and in the direction θ from a small scatterer of mass M is from Equation (35)

$$\Phi_2(r_2) = \Phi_1(r_1)\, M r_2^{-2} \left[0.024\, K(E, \theta)\right] \text{ gammas } cm^{-2} \, sec^{-1} \qquad (36)$$

The function $K(E, \theta)$ is plotted in Figure 23. The quantity $0.024\ K(E, \theta)$ may be regarded as the probability that the photon energy will scatter into unit solid angle at the angle θ, per unit mass of scatterer.

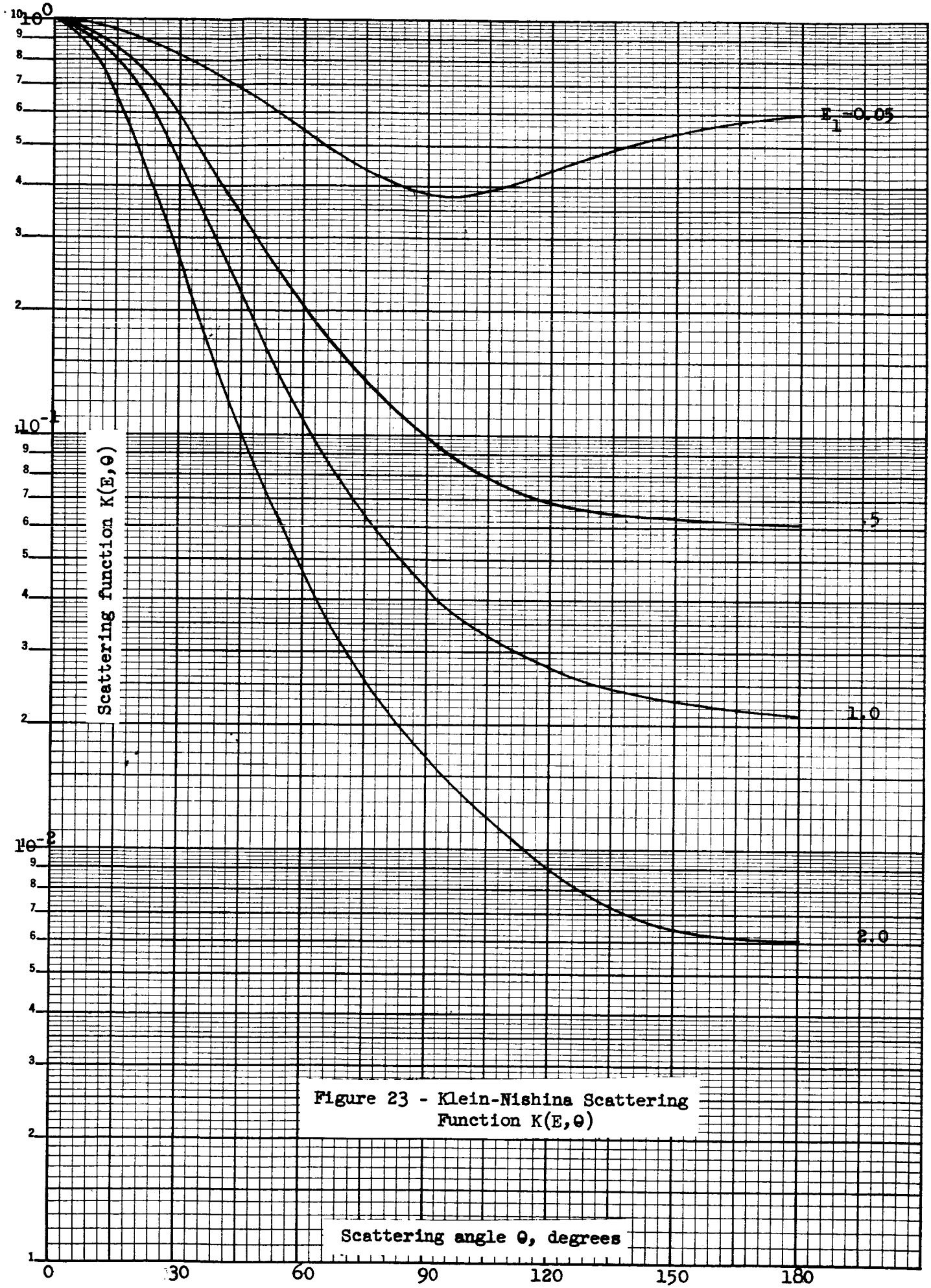


Figure 23 - Klein-Nishina Scattering Function K(E,θ)

3. 6. 1. 2 Thick Scatterers

A thick scatterer differs from the above case in that absorption in the scatterer must be taken into account. This may be calculated as follows: Let

θ_1 = angle between the incident beam and the normal to the surface of the slab

θ_2 = angle between the emergent beam and the normal to the surface of the slab

θ = angle of scattering ($180^o - \theta_1 - \theta_2$)

x = distance below the surface of the slab (measured along the normal to the surface) at which scattering occurs

dx = element of thickness of slab

Σ_1, Σ_2 = absorption coefficients of the slab for incident and emergent radiation, respectively

$\Phi_1(r_1)$ = intensity of the beam arriving at the surface of the slab

$\Phi_2(r_0)$ = intensity 1 cm from the surface of the slab

A = cross sectional area of the incident beam if the slab is larger than the beam; or, the projected area of the scatterer in the direction of the incident beam if the beam is larger than the scatterer

The value of $\Phi_2(r_0)$, the intensity at distance 1 cm from the slab in the direction of the observer is the integral of the following factors:

$\Phi_1(r_1)$ = intensity of beam arriving at the slab

$e^{-\Sigma_1 x \sec \theta_1}$ = fraction of incident energy which penetrates to a depth x

$\rho A \sec \theta_1 dx$ = mass of scattering element of area A sec θ_1 thickness dx, and density ρ.

$0.024\ K(E, \theta)$ = probability of scattering into unit solid angle in direction θ_2, per unit mass of scatterer

$e^{-\Sigma_2 x \sec \theta_2}$ = probability of escape of scattered photon through the surface of the slab

$$\Phi_2(r_0) = \Phi_1(r_1) A \int_0^\infty \rho\ 0.024\ K(E, \theta)\ \sec \theta_1\ e^{-(\Sigma_1 \sec \theta_1 + \Sigma_2 \sec \theta_2)X} dx \quad (37)$$

which has the following solution

$$\frac{\Phi_2(r_0)}{A\Phi_1(r_1)} = \frac{0.024\,\rho\,K(E,\theta)}{\Sigma_1 + \dfrac{\Sigma_2 \cos\theta_1}{\cos\theta_2}}$$

$$= \frac{0.024\,K(E,\theta)}{\dfrac{\Sigma_1}{\rho} + \dfrac{\Sigma_2}{\rho}\dfrac{\cos\theta_1}{\cos\theta_2}} \tag{38}$$

$$= R(E_1\ \theta_1\ \theta_2)$$

When only Compton scattering is involved (slab contains only light elements and gamma energy is high), the mass absorption coefficients Σ/ρ are functions of energy only and are independent of the composition of the slab. Under these conditions the function $R(E_1\,\theta_1\,\theta_2)$ which is plotted in Figure 24 may be used as a reflection coefficient, since it is essentially the ratio

$$\frac{\text{energy scattered into unit solid angle toward the observer}}{\text{total energy arriving at the slab}}$$

When the geometry is such that both source and scatterer may be treated as points, the intensity at the observer is

$$\Phi_2(r_2) = \Phi_1(r_1)\,A\,r_2^{-2}\,R(E_1\ \theta_1\ \theta_2) \tag{39}$$

3. 6. 2 Neutron Scattering

3. 6. 2. 1 Structure Scattering

The structure scattering for neutrons may be calculated in a manner similar to that for gamma ray flux as given in Equation

$$\Phi_2(r_2) = \Phi_1(r_1)\,\Delta A\ r_2^{-2}\,R_n\,(E_1\ \theta_1\ \theta_2) \tag{40}$$

This equation assumes that the scattering elements ΔA are sufficiently small and r_1 and r_2 sufficiently large that the scatterer may be considered a point source. Unfortunately reflection coefficients $R_n(E_1\ \theta_1\ \theta_2)$ are not available in the same form as that given for the gamma photons. Another factor R(T) called the dose albedo for neutrons as a function of the thickness T of the scattering medium, however, has been reported.*

*Reactor Handbook, Second Edition, Vol. III, Part B, page 281.

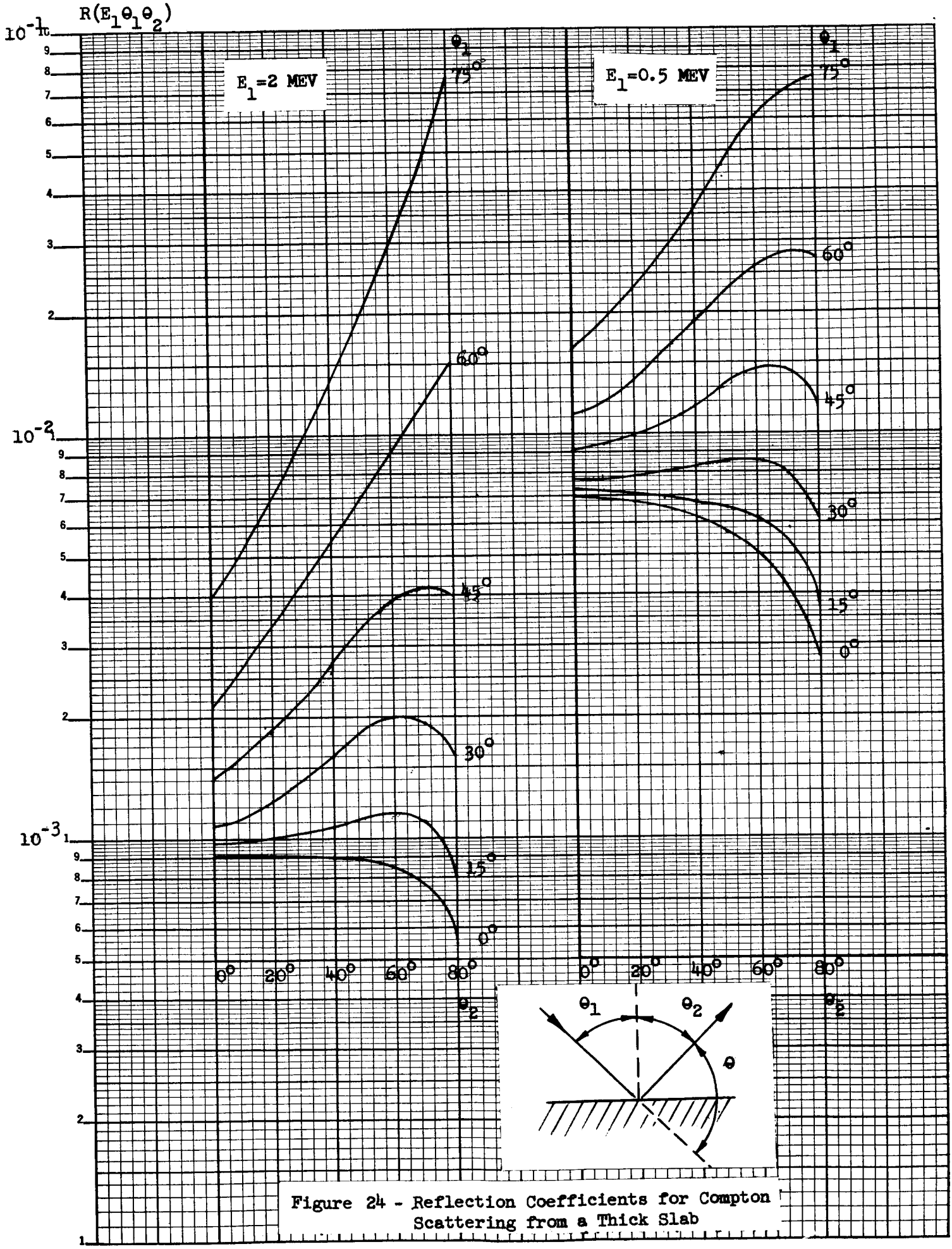


Figure 24 - Reflection Coefficients for Compton Scattering from a Thick Slab

where $\Phi(\theta_0, g)$ = neutron flux at observer located at g feet from the source for a beam of monoenergetic neutrons leaving the source at an angle θ_0 with the source-observer line

θ_0 = angle of source beam with source-receiver line

g = separation distance in feet (calculations include distances up to 100 feet)

$\Sigma_S(E_0)$ = macroscopic scattering cross section in air at initial energy E_0

3. 6. 2. 3 Shield-Surface Scattering

In the design of shadow shields for space applications the knowledge of the angular distribution of the radiation leaving the surface of the shield becomes important.

As an example consider the sketch in Figure 25 where a reactor and a detector are located in some fixed geometry. Assuming that the surface or diffuse components of the radiation leaving the surface of the shield play an important role in the amount of low and intermediate energy neutrons that are detected by a neutron dosimeter, then a comparison may be made for a (1) slab shadow shield and (2) a proposed conical shield as shown in the figure. The object of a conical design is to direct the diffuse component of the radiation away from the detector position rather than toward it. The collimated component, however, will not be appreciably different for either design. This analysis is based on the case where the detector is located twice as far from a shield surface as the shield is from the reactor. If the half angle of the cone (α) is varied but the axial thickness T_0 of the shield is fixed, then assuming in this ideal study that the amount of structure does not increase significantly for reasonable cone angles, the shield weight for the two configurations should be essentially the same. The objective of this study, therefore, is to demonstrate the decrease in the diffuse component of the radiation at some detector position for a fixed shadow shield weight system.

Figure 26 illustrates the importance of the angular distribution of the radiation leaving the surface of the shield. In this figure, the dose due to the diffuse radiation at the detector position for the flat shield divided by the dose calculated at the same position for the conical shield of half angle alpha is shown as a function of alpha. For 90^o this ratio should be, and is, equal to unity. It may be seen that as the angular distribution of the radiation becomes more forward on the surface of the shield the gain increases. For the case of the cosine of the first power it may be seen that the change in dose rate does not vary too much from unity. This is

For the case of iron, lead and lucite it has been shown that for T = 1/2 inch the albedo R(T) is equal to 0. 10 for the dose of neutrons which were incident normally to the slab. For larger thicknesses (about 4. 5 inches) the iron and lead reflects about 50 percent of the neutron dose and the lucite about 25 percent.

An approximate calculation of scattered neutrons may be performed as follows: Let

$\Phi_1(r_1)$ = number of neutrons arriving at a unit mass of scattering material
Σ_s = macroscopic scattering cross section, cm^{-1}
ρ = density of the material, gm cm^{-3}
ΔM = mass of scattering element
$\Phi_2(r_2)$ = number of neutrons arriving at the observer
r_1, r_2 = distance of source of neutrons to the scattering mass and distance from the scatterer to the observer

By assuming isotropic scattering and no energy loss on scattering or absorption in other material the following equation may be written

$$\Phi_2(r_2) = \Phi_1(r_1) \frac{\Sigma_s/\rho}{4\pi r_2^2} \Delta M \quad \text{neutrons cm}^{-2} \text{ sec}^{-1} \tag{41}$$

where $\Phi_1(r_1) = \dfrac{Q_P}{4\pi r_1^2}$

and Q_P = point isotropic source, neutrons sec^{-1}

3. 6. 2. 2 Air Scattering

Air scattering of neutrons has played an important role in the ANP program where the divided shield system depended on accurate predictions of the scattered neutrons arriving at the crew compartment. Consequently a large amount of computations have been made and reported in the references listed.

It has been shown* by Monte Carlo calculations that multiple scattered neutrons in air may be approximated by

$$\Phi(\theta_o, g) = \frac{\Sigma_s(E_o)}{4\pi g} \frac{(\pi - \theta_o)}{\sin\theta_o} \frac{\text{neutrons cm}^{-2} \text{ sec}^{-1}}{\text{source neutron sec}^{-1}} \tag{42}$$

*F. L. Keller, et al. Monte Carlo Calculations of Fluxes and Dose Rates Resulting from Neutrons Multiply Scattered in Air. ORNL-2375 (1958).

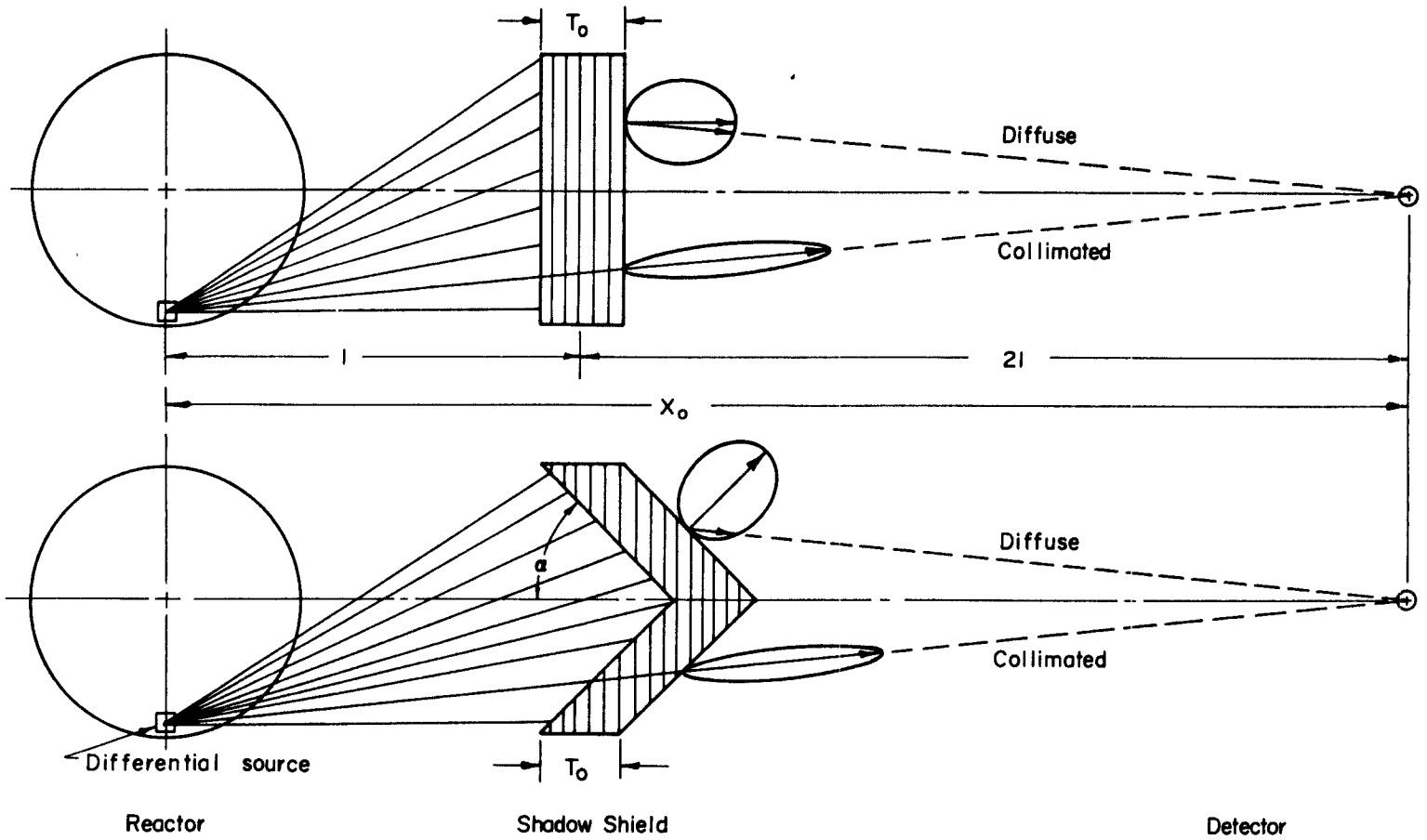


Figure 25 - Comparison of Two Possible Types of Shadow Shield Configurations

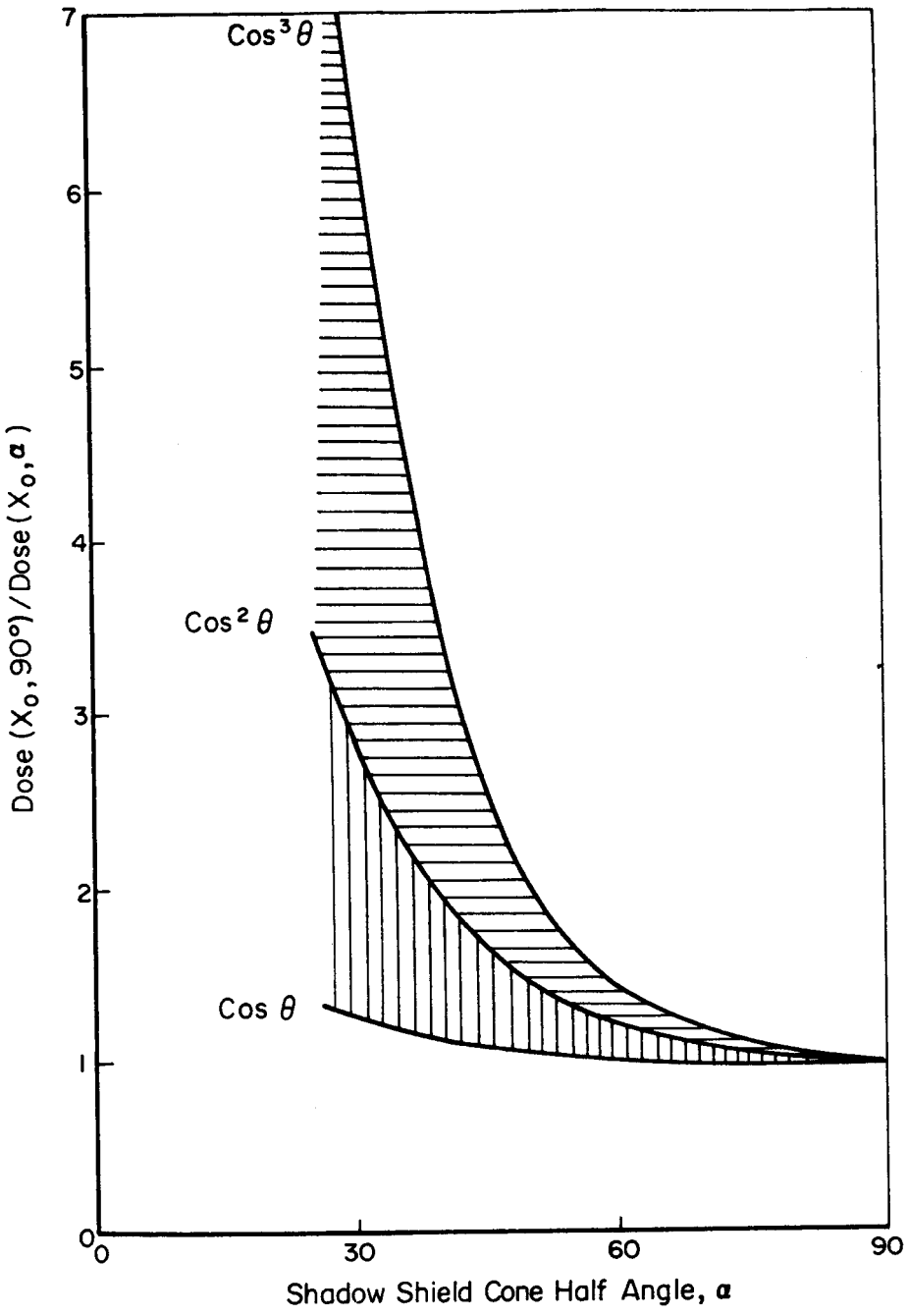


Figure 26 - Diffuse Component Dose Reduction Factor For Various Surface Angular Distributions as a Function of C one Half Angle

due to the compensating effect of shield surface area increasing as the cosine of the first power. The slight increase noted below 30 degrees for this case is a geometry effect. It is felt that for the diffuse component, the angular distribution leaving the surface of the shield should at least be in the neighborhood of cosine squared theta and, therefore, for a co-angle of about 40 degrees a factor of two decrease in the diffuse dose may be obtained by considering the conical design compared to that of a slab design. Since in the many cases studied, the collimated component at equilibrium shield thicknesses was found to contribute only about 10 percent of the dose the observed reduction of a factor of two would, therefore, be applied to 90 percent of the total dose of this system.